Project FORTE - Nuclear Thermal Hydraulics Research & Development

Review of Thermal Hydraulic Test Facilities

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FRAZER-NASH CONSULTANCY

SYSTEMS AND ENGINEERING TECHNOLOGY

An introduction to Project FORTE

The Department for Business, Energy and Industrial Strategy (BEIS) has tasked Frazer-Nash Consultancy and its partner organisations to deliver the first phase of a programme of nuclear thermal hydraulics research and development.

Phase 1 of the programme comprises two parts:

- The specification and development of innovative thermal hydraulic modelling methods and tools; and
- The specification of a new United Kingdom thermal hydraulics test facility.

The work is intended to consider all future reactor technologies including Gen III+, small modular reactors and advanced reactor technologies.

Our project partners

The team is led by Frazer-Nash Consultancy and includes:

FRAZER-NASH CONSULTANCY

The University Of Sheffield.

EDF ENERGY

Westinghouse

MANCHESTER 1824
The University of Manchester

Science & Technology Facilities Council

For more information, visit www.innovationfornuclear.co.uk/nuclearthermalhydraulics.html

Executive Summary

A UK National Nuclear Thermal Hydraulic Test Facility will provide a much-needed resource to support the UK nuclear industry. However, it is also intended to target some of the gaps between the global state-of-the-art and the capability that industry needs, thereby making a positive and distinctive worldwide contribution.

Therefore, a definitive review of the state-of-the-art of worldwide thermal hydraulic research facilities with relevance to Small Modular Reactor and Generation IV nuclear reactor designs has been undertaken. This includes thermal hydraulic test facilities associated with molten salt, lead, sodium, gas and water cooled reactor designs.

This report provides a summary of the thermal hydraulic test facilities organised by country, and highlights some of the key parameters associated with each test facility. This demonstrates that significant experimental testing has been conducted across all Generation III, Small Modular Reactor and Generation IV reactor technologies throughout the world, particularly in Canada, China, France, Germany, India, Italy, Japan, Korea, Switzerland, Russia and USA.

Although this report does not provide a definitive, exhaustive list, the number of test facilities for each reactor technology gives an indication of the extent of the international research:

- 87 water (PWR, BWR and SCWR) reactor Separate Effect Test facilities for natural circulation, two-phase flow, flow-induced vibration and component flow phenomena;
- 34 water (PWR and BWR) reactor containment facilities for combustion, condensation, aerosol and non-condensable gas behaviour;
- 24 gas (HTGR, VHTR and GFR) thermal hydraulic test facilities;
- 42 sodium (SFR) thermal hydraulic test facilities;
- 31 lead (LFR) thermal hydraulic test facilities; and
- 16 molten salt (MSR) thermal hydraulic test facilities.

This review demonstrates that there are a large number of successful nuclear thermal hydraulic test facilities around the world with comprehensive coverage of phenomena and reactor types. Therefore, any new facility will need to be carefully planned in order to make a major impact.

Test facilities in a few organisations have been reviewed in more detail with a focus on ‘facility’ capability rather than test rig capability, in order to better understand the current state-of-the-art in thermal hydraulic test facilities. These examples highlight how existing test facilities have developed a strong, distinctive capability within nuclear thermal hydraulics with their own individual areas of research, measurement techniques or specific expertise.

In the past, experimental facilities have generally focused on tests that resulted in data more suitable for safety cases and the validation of system codes. However, current reactor developments require more extensive validation data to develop more efficient and less over-engineered designs in shorter development periods. This requires more detailed data from extensive experimental datasets and the use of advanced diagnostic techniques.

The number and breadth of experimental facilities that are currently built and operated around the world shows that the international nuclear community recognises the need for experimental research and testing to support nuclear reactor development. There is consequently room for the UK to play a part in meeting these needs. However, in order for the UK facility to be successful, it needs to develop its own unique expertise and distinctiveness. One strong possibility is in high quality novel measurement and diagnostic techniques to generate data of sufficient quality to be used for CFD model validation.

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1 Introduction

1.1 Project Context

The UK Government's 2013 Nuclear Industrial Strategy described significant ambitions for the UK to grow our nuclear capability; with the key aim of becoming the preferred nation state partner of the global nuclear technology industry. To help fulfil the strategy's initial objectives, the Nuclear Innovation and Research Advisory Board (NIRAB) was established in 2014. NIRAB comprised experts from both industry and academia with the objective of advising Government on the approach to and coordination of nuclear innovation, research and development in the UK. In March 2016, NIRAB published a set of recommendations for innovation and research programmes [1], the recommendations being prioritised in a subsequent report [2].

In addition, the 2018 Nuclear Sector Deal between the UK Government and the nuclear industry [3] demonstrates the Government's commitment to nuclear research and initial funding for the Advanced Modular Reactor (AMR) initiative.

Control of thermal hydraulic phenomena is at the heart of all current and future nuclear reactor designs, underpinning both reactor performance and safety arguments. Therefore, a thorough understanding of thermal hydraulic effects, the capability to simulate them accurately and validate the key prediction methodology experimentally is essential for efficient design and safe operation through life.

Historically the UK has had a strong capability in thermal hydraulic science and engineering, which was derived from and could be applied across many industries. Several of the most successful commercial Computational Fluid Dynamics (CFD) codes have their origins within UK research institutions. There remains considerable UK-based expertise and impetus in the development of thermal hydraulic modelling tools and techniques to service the needs of other industries. However, the general lull in activity in the UK civil nuclear industry since Sizewell B was commissioned in 1995, and the cancellation of the Dounreay fast reactor programme, is also evident in nuclear thermal hydraulic research. The UK currently has no major civil nuclear thermal hydraulic test facilities and activity in modelling R&D has been limited by a lack of funding or other stimulus, from either the UK industrial base or government.

In order to begin to arrest this trend, and within the overall aim of allowing nuclear energy to play a significant role in the UK's future energy mix, two of the NIRAB recommendations relate to nuclear thermal hydraulics:

- The development of a major new UK Nuclear Thermal Hydraulic Test Facility;
- The development of new Nuclear Thermal Hydraulic Modelling Techniques and Tools.

1.2 Programme Goals

Quoting from the ITT for this work [4], the long term goals are:

- By 2020 establish the UK as a partner engaged in collaborative design projects for new reactors (Generation IV and SMR), building on its existing and growing design expertise.
- By 2030 maturing R&D results in deployment of new plant with significant UK design content and manufactured parts.
- By 2050 R&D has facilitated UK industry to be a significant partner in the global deployment of Gen III+, Gen IV and Small Modular Reactor (SMR) technologies.

The specific benefits of the five-year programme are expected to be:

- Enhanced designs, increased productivity and a step change in the way that nuclear design, development and construction programmes are delivered;
- Increased and widespread uptake of modern digital engineering practices within the UK nuclear industry;
- Improved understanding and safety of through life performance of reactor components;
- A greater predictive modelling capability and understanding of passive safety arguments;
- A highly-skilled workforce able to drive design improvements and underpin operations and regulation of future reactors; and
- Leverage to facilitate extended UK participation in associated international activities.

All of these aims and benefits cannot be met by nuclear thermal hydraulic research alone, however, all activities and outputs from this project have been considered in the light of whether they move the programme in the desired direction as well as whether they achieve the specific project deliverables.

1.3 Report Structure and Objectives

In order to understand the requirements for the UK National Nuclear Thermal Hydraulic Test Facility, it is first necessary to understand and assess the existing experimental thermal hydraulic test facilities and current state-of-the-art around the world. This will feed into the test facility specification and ensure that the UK test facility will support worldwide thermal hydraulics development and improve UK international collaboration.

This review is one part of the work being undertaken within Project FORTE to support the development of a UK National Nuclear Thermal Hydraulic Facility, and identifies thermal hydraulic test rigs and facilities around the world of relevance to SMR and Gen IV reactor designs.

The objective of this review is to highlight the state-of-the-art in nuclear thermal hydraulics testing, and understand the current research focus, number of test rigs and instrumentation and measurement capability within existing worldwide test facilities. This includes a review of a number of nuclear thermal hydraulic test facilities around the world with an emphasis on ‘facility’ capability rather than test rig capability.

The structure of this document is as follows:

- A summary of the Gen III+, SMR and advanced (Gen IV) reactor technologies under development is given in Section 2.
- General requirements for nuclear thermal hydraulic test facilities, classification of test rigs and summary of relevant quality assurance accreditations are discussed in Section 3.
- A high level description of the measurement techniques that are currently used in nuclear thermal hydraulics test rigs is provided in Section 4.
- Sections 5, 6, 7, 8 and 9 provide a detailed review of selected test facilities that focus on water, gas, sodium, lead and molten salt reactor coolant technologies respectively.
- Finally, Section 10 summarises the conclusions from the review of experimental test rigs.

2 Current Reactor Development

This section summarises the current nuclear reactor development that is being undertaken around the world associated with advanced (Gen IV) reactors (Section 2.1) and SMR designs (Section 2.2). A summary of the reactors under development around the world is provided in Section 2.3 and listed in Annex B.

2.1 Gen IV Reactors

Since 2000, the Generation IV International Forum (GIF)[1] has been a focus for international collaborative efforts to develop next generation nuclear energy systems that can help meet the world's future energy needs. Gen IV designs are expected to be more fuel efficient, reduce waste, improve safety and reduce cost. With these goals in mind, the GIF selected the following six reactor technologies for further research and development:

- Gas-cooled fast reactor (GFR);
- Very-high-temperature reactor (VHTR);
- Sodium-cooled fast reactor (SFR);
- Lead-cooled fast reactor (LFR);
- Molten salt reactor (MSR); and
- Supercritical-water-cooled reactor (SCWR).

The main characteristics and timeline for each of the six Gen IV reactor technologies is described in the Gen IV 2014 technology roadmap [5].

2.2 Small Modular Reactors

There is strong interest in small and simpler units for generating electricity from nuclear power, and for process heat. This interest in small and medium sized nuclear power reactors is driven both by a desire to reduce the impact of capital costs, factory fabricate significant portions of the reactor and to provide power for smaller grid systems, or away from large grid systems. The technologies involved are numerous and diverse.

The International Atomic Energy Agency (IAEA) defines 'small' reactors as under 300 MWe, and up to about 700 MWe as 'medium', including many operational units from 20th century. Together they are now referred to by the IAEA as small and medium reactors. However, 'SMR' is used more commonly as an acronym for 'small modular reactor'.

2.3 Reactors Under Development

The current Gen IV and SMR reactors under development have been taken from the IAEA Advanced Reactor Information System (ARIS) database[2], ARIS publications [6] and the World Nuclear Association website[3]. This provides a compilation of reactor technology developments tabulated according to coolant technology and organised by country.

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[1] https://www.gen-4.org
[2] https://aris.iaea.org
[3] http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors.aspx

For each reactor design, the organisation, country, coolant and electrical power is listed, and categorised as follows in Annex B:

- Table 10: Pressurised Water Reactor Designs
- Table 11: Integral Pressurised Water Reactor Designs
- Table 12: Boiling Water Reactor Designs
- Table 13: Heavy Water Reactor Designs
- Table 14: SuperCritical Water Reactor Designs
- Table 15: Gas-cooled Reactor Designs
- Table 16: Sodium-cooled Fast Reactor Designs
- Table 17: Lead-cooled Fast Reactor Designs
- Table 18: Molten Salt Reactor Designs

3 Nuclear Thermal Hydraulics Test Facility

A nuclear thermal hydraulics test facility is considered to be one or more buildings that contain experimental test rigs, which are designed to simulate nuclear thermal hydraulic phenomena. Test facilities are managed and operated around the world by academic and research institutions, as well as national laboratories, commercial organisations and reactor developers, to perform the following functions:

- Improved understanding and investigation of nuclear thermal hydraulic phenomena.
- Measurement of thermal hydraulic parameters, such as velocity, pressure, temperature and void fraction for the development of correlations and closures for system and sub-channel codes and the validation of Computational Fluid Dynamics (CFD) models.
- Testing and qualification of systems and components under reactor conditions to confirm the performance of key reactor components for example pumps and valves.

Thermal hydraulic test rigs within a facility are designed with a specific purpose and type of tests that will be undertaken. Thermal hydraulic tests are often classified as basic, separate effect or integral effect tests, which are discussed in more detail in Section 3.1.

The quality, accuracy and repeatability of experimental test data is essential to satisfy nuclear industry regulation requirements. This is particularly important for testing and qualification of nuclear components, but is also necessary for experimental data that are used to support a nuclear safety case or validate a model that is used for safety justification or reactor design. The main quality assurance accreditations for nuclear thermal hydraulic test facilities are discussed in Section 3.2.

In addition, the quality and type of instrumentation and measurement (including data acquisition and reduction) that is available within a test facility and installed on particular test rigs has a large impact on the benefit and value of a test rig and the data that it generates. The techniques that are currently used to measure thermal hydraulic parameters are discussed in Section 4.

3.1 Classification of Test Rigs

Thermal hydraulics has always played a central role in the design, safety assessment, operation, and maintenance of nuclear reactor designs. Experiments were used before the widespread availability of computers to estimate, understand, and prepare models of thermal hydraulic phenomena that may appear in NPPs. Therefore, experiments have formed the basis of nuclear thermal hydraulics to meet the safety evaluation requirements and have underpinned the development of computational tools. Currently, there is a tight connection between experiments and the development and qualification/validation of computer codes.

As discussed in NEA/CSNI/R(2016)14 [7], experiments can be classified into three categories:

- Basic tests;
- Separate Effect Tests (SETs); and
- Integral Effect Tests (IETs).

Although these categories of experiments were developed for LWRs, they are applicable to all reactor types and are explained briefly in Sections 3.1.1 to 3.1.3.

3.1.1 Basic Tests

This category of experiments addresses fundamental phenomena, such as pressure drops, single and two-phase flows, fluid mixing, heat transfer, including boiling and condensation, critical flow, pressure-wave propagation, and complex phenomena due to combinations of fundamental processes like flooding and counter-current flow limitation.

Basic tests aim to understand the phenomena under simple, steady-state boundary conditions, sometimes with less reference to actual reactor conditions, including those expected in accidents. Rather, basic tests may reveal information essential for developing models and provide data that can be used to improve confidence in modelling tools.

3.1.2 Separate Effect Tests

Validation of codes and models is more tractable when local phenomena are separated from the whole system response where various phenomena interact. Separate effect tests vary in scale, potentially addressing multiple phenomena occurring in reactor components and sub-assemblies. They are typically performed on individual, representative test pieces under more complicated boundary conditions than basic tests. Separate effects tests may provide data that can be used to improve confidence in more complex 3-D models. For example, local phenomena suitable for SET are expected to occur in and around the following:

- Primary thermal hydraulic regions or zones of reactors; and
- Components, such as centrifugal pumps, valves, separators, dryers, jet-pumps, accumulators.

Scaling of equipment (by height, volume, Reynolds number, Grashoff number etc) is important for separate effect tests [7] to ensure that the thermal hydraulic phenomena of interest is representative of the full-scale reactor conditions.

3.1.3 Integral Effect Tests

Integral Effect Test (IET) facilities are designed and operated to reproduce aspects of a reference reactor's performance and behaviour. IET facilities are large test facilities that address the performance of an entire reactor system in support of a particular reactor development programme that are performed on a scaled representation of the primary and/or secondary circuit(s) under reactor transient and accident conditions. Integral effects tests also provide data to improve confidence in predictive models and substantiate the design.

Scaling is essential for designing and operating IETs by considering the thermal hydraulic phenomena of interest and aspects of the reactor performance that are being represented in the test programme. The scaling of nuclear thermal hydraulic experimental tests is a whole topic on its own, which is covered in detail in NEA/CSNI/R(2016)14 [7].

3.2 Quality Assurance

The safety standards issued by the International Atomic Energy Agency (IAEA)[4] reflect an international consensus on what constitutes a high level of safety for protecting people and the environment. They often serve as the basis for nuclear safety regulatory documents, and are incorporated in Member States' national nuclear regulations.

In order to apply these standards effectively, the IAEA safety standards need to be complemented by industry standards, and must be implemented within an appropriate national regulatory infrastructure, such as the Office for Nuclear Regulation (ONR) in the UK.

In addition, nuclear laboratories and test facilities are often accredited to a number of international industry standards to demonstrate and ensure that appropriate quality assurance and test procedures are in place and operated by the test facility.

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[4] http://www-ns.iaea.org/standards/documents/pubdoc-list.asp?s=11&l=96

This section considers the Quality Assurance (QA) requirements and accreditations for a nuclear thermal hydraulic test facility, although this cannot be considered a definitive list. This is broken down into quality management systems (Section 3.2.1), testing and calibration (Section 3.2.2) and nuclear quality requirements (Section 3.2.3).

3.2.1 Quality Management Systems

ISO 9001 is the international standard that specifies requirements for a Quality Management System (QMS) [8]. Organisations use the standard to demonstrate the ability to consistently provide products and services that meet customer and regulatory requirements. ISO 9001 certification ensures organisations continually improve their processes and manage key business risks. For example, Lloyd's Register provides UKAS-accredited ISO 9001 certification, training and gap analysis services in the UK.

NSQ-100 is a standard that has been developed by the Nuclear Quality Standard Association (NQSA) that is dedicated to the quality of the nuclear supply chain [9]. This standard is based upon the principles of ISO 9001:2008 and two other major nuclear quality standards: IAEA Safety Standards Series No. GS-R-3:2006 and ASME NQA-1-2008 and addenda 2009.

ISO 19443 (Substitute for NSG-100 standard) details the application of ISO 9001:2015 to organisations supplying products or services important to nuclear safety [10]. It specifies requirements for a quality management system when an organization: a) needs to demonstrate its ability to consistently provide products and services that meet customer and applicable statutory and regulatory requirements, and b) aims to enhance customer satisfaction through the effective application of the system, including processes for improvement of the system and the assurance of conformity to customer and applicable statutory and regulatory requirements.

ISO 45001 specifies requirements for an Occupational Health and Safety (OH&S) management system, and gives guidance for its use, to enable organisations to provide safe and healthy workplaces by preventing work-related injury and ill health, as well as by proactively improving its OH&S performance [11]. It is applicable to any organisation that wishes to improve occupational health and safety, eliminate hazards and minimize health and safety risks (including system deficiencies). ISO 45001 is a replacement for BS OHSAS 18001.

3.2.2 Testing and Calibration

ISO 17025 specifies the general requirements for the competence to carry out tests and/or calibrations, including sampling [12]. It covers testing and calibration performed using standard methods, non-standard methods, and laboratory-developed methods. It is applicable to all organizations performing tests and/or calibrations. This includes laboratories where testing and/or calibration forms part of inspection and product certification. It is for use by laboratories in developing their management system for quality, administrative and technical operations. Laboratory customers, regulatory authorities and accreditation bodies may also use it in confirming or recognizing the competence of laboratories.

ISO 17020 specifies requirements for the competence of bodies performing inspection and for the impartiality and consistency of their inspection [13]. It covers the activities of inspection bodies whose work can include the examination of materials, products, installations, plants, processes, work procedures or services, and the determination of their conformity with requirements and the subsequent reporting of results of these activities to clients and, when required, to authorities. Inspection can concern all stages during the lifetime of these items, including the design stage. Such work normally requires the exercise of professional judgement in performing inspection, in particular when assessing conformity with general requirements.

United Kingdom Accreditation Service (UKAS) is the sole national accreditation body for the United Kingdom. UKAS is recognised by government, to assess against internationally agreed standards, organisations that provide certification, testing, inspection and calibration services, such as ISO 17020 and ISO 17025. UKAS accreditation provides an assurance of the competence, impartiality and integrity of conformity assessment bodies. UKAS' involvement in international groups such as the International Laboratory Accreditation Cooperation (ILAC), provides for mutual recognition. This means that UKAS accreditation to ISO 17025 has worldwide cross acceptance through ILAC to:

- ANSI-ASQ National Accreditation Board (ACLASS), USA;
- China National Accreditation Service for Conformity Assessment (CNAS), China;
- Comite Francais d'Accreditation (COFRAC), France;
- Deutsche Akkreditierungsgesellschaft (DAkkS), Germany;
- National Accreditation Board for Testing & Calibration (NABL), India; and
- Entidad Nacional de Acreditacion (ENAC), Spain.

3.2.3 Nuclear Quality Requirements

Quality Assurance requirements provide control/regulatory bodies with confidence and assurance that people, processes, and implementation will meet standards and performance expectations. These requirements can be different depending on the standards of the IAEA member states control/regulatory bodies. Therefore, the quality requirements for the test facility will depend on the member state control/regulatory body that the test or equipment qualification is intended to satisfy. For example in the US:

ASME NQA-1 (with related addenda) defines the quality assurance requirements for nuclear facility applications [14], and was developed by the American Society of Mechanical Engineers (ASME). The basic QA requirements that need to be complied with when running thermal hydraulic experiments, to make sure that data will be accepted by ASME, is stated in Requirement 11: Test Control [14]. By adhering strictly to QA, risk is reduced by a process of verifying the majority of or all aspects of an activity: Plan or design; Materials used; Attention to detail during construction; Thoroughness of testing; Conduct of maintenance and operations; Control of procurement spare parts; and Document control.

10 CFR 50 Appendix B Quality Assurance Criteria is applied to the design, fabrication, construction, and testing of the structures, systems, and components of the nuclear power plants [15]. This Code is written by the US Nuclear Regulatory Commission (NRC) and has 18 Criteria. Criteria 11 - Test Control states that:

> *'A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.'*

4 Measurement Techniques

An essential aspect of all nuclear thermal hydraulic test rigs is the accuracy, resolution and type of measurements that are made. The decision depends on the requirements and objectives of each individual test programme, which could be to advance current understanding of thermal hydraulic behaviour, provide high quality data for validation and verification of CFD models and system codes or compile databases of experimental test data to support the development of empirical correlations.

Suitable and reliable instrumentation is a necessity that bridges the gap between experiment and analysis. Instruments must be able to operate adequately in the specific test conditions they will be subject to, and so some instruments must be capable of withstanding high pressures, temperatures and corrosive environments. In these conditions, the instrument must also provide an output signal that honours the fidelity of the physical parameter being measured. As such, many instruments are not suitable for general measurement techniques, but can only be reliably applied over certain parameter ranges. Many different instruments therefore exist for obtaining experimental data.

This section summarises some of the instrumentation and measurement techniques that are currently used to measure thermal hydraulic parameters. These measurement techniques are grouped according to the parameter being measured, as follows:

- Flow rate (Section 4.1)
- Pressure (Section 4.2)
- Temperature (Section 4.3)
- Velocity (Section 4.4)
- Flow visualisation (Section 4.5)
- Void fraction (Section 4.6)
- Film thickness (Section 4.7)
- Vibration (Section 4.8)

4.1 Flow Rate

4.1.1 Orifice Plate

An orifice plate is a thin metal disc containing a concentric hole, which is inserted into the pipe carrying the flowing fluid (Chapter 12 in [16]). Flow is restricted by the hole which introduces a differential pressure across the orifice. The upstream and downstream pressures are measured using a differential pressure gauge (Section 4.2.1), and a calibrated loss coefficient is applied to calculate the flow rate. Orifice plates are a simple, cheap and widely used method for measuring volumetric flow rate. However, a limitation of this method is that measurement inaccuracy is typically ±2%, but can reach as high as ±5% (Chapter 12 in [16]).

4.1.2 Coriolis Flow Meter

Coriolis flow meters are typically used to measure the mass flow rate of liquids, but can also be used on some gases and can be applied to indirectly measure volumetric flow rate. A Coriolis flow meter contains a pair of parallel vibrating tubes, or a single vibrating tube formed into two parallel sections (Chapter 16 in [17]). As fluid flows through the vibrating tube(s), a Coriolis force is generated and the tube is deflected further to the existing vibratory motion. This deflection is proportional to the mass flow rate of the fluid, which can be measured using a suitable sensor. Coriolis flow meters have a high accuracy of ±0.2% (Chapter 16 in [17]) and can be used on liquids, gases, slurries and two-phase flows (Chapter 12 in [16]). However, they are expensive and can suffer from mechanical problems such as fatigue and corrosion.

4.1.3 Doppler Shift Ultrasonic Flow Meter

A Doppler shift flow meter consists of an ultrasonic transmitter and receiver which are clamped to the outside of a pipe or fluid carrying vessel (Chapter 16 in [17]). The transmitter emits ultrasonic waves which are deflected by scattering elements in the fluid and received by the receiver. This deflection causes a change in wave frequency, which is used to determine the flow rate. The Doppler shift flow meter is inexpensive and can be used on gases or liquids; however the measurement accuracy depends on a number of different parameters, and so accurate measurements require careful calibration.

These instruments are typically used for flow indications instead of accurate quantification of volumetric flow rate (Chapter 16 in [17]). Variants with higher measurement accuracy have been developed, however these can be significantly more expensive. However, since these flow meters are clamped to the outside of a pipe, no contact is required between the flow meter and fluid being measured.

4.2 Pressure

4.2.1 Pressure Transducer

Pressure can be quantified in three ways (Chapter 15 in [17]): absolute pressure (a), gauge pressure (g) or differential pressure (d). The most common way of monitoring wall-static pressure is via pressure taps, which are small holes drilled perpendicular to the surface. To measure the pressure, long thin tubes connect the taps to a pressure measuring device located outside the test rig, such as a pressure transducer or manometer.

Pressure transducers typically contain an elastic-element that deforms under the applied static pressure, such as a diaphragm, bellows or Bourdon tube. This displacement is then measured by the transducer and converted to an electrical signal which is used to determine the pressure. For most applications, a diaphragm pressure transducer is normally used.

Pressure transducer diaphragms are fabricated from different materials so that they can be suited to specific environments e.g. plastic, stainless steel, ceramic or silicon sheet. These different types vary in measurement accuracy and cost, and can be used to measure absolute, differential or gauge pressure.

Electronic pressure gauges use different methods for measuring the displacement in the diaphragm to improve performance, such as piezoresistive, piezoelectric, resonant, magnetic, capacitive and fibre optic sensors. For example, monolithic piezoresistive pressure transducers with the signal processed using an active linearization circuit can improve the measurement accuracy to as low as ±0.1% of the full scale reading (Chapter 15 in [17]).

4.2.2 Pressure Sensitive Paint

Pressure Sensitive Paint (PSP) is a method for measuring air pressure or local oxygen concentration, usually in aerodynamic settings [18]. PSP is a paint-like coating which fluoresces under a specific illumination wavelength in differing intensities depending on the external air pressure being applied locally to its surface. PSP provides a low-cost alternative that is less invasive than pressure tap arrays. PSP also offers superior spatial resolution, with each pixel of the imaging camera acting as a pressure tap. PSP can achieve accuracy within 150 Pa of pressure tap measurements with good setup and experience.

Time-resolved PSP applications involve pulsed excitation and delay and gating of the imaging devices. One can thus determine pressure differentials as a function of time. In this case, the imaging devices must be synchronized to the excitation with multi-channel digital delay/pulse generators providing that synchronization. This technique is used in aerodynamic and turbomachinery applications.

Since PSP uses imaging cameras, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.2.3 Fibre-Optic Pressure Sensor

One form of a fibre-optic pressure sensor is known as a micro-bend sensor (Chapter 15 in [17]). In this, the refractive index of the fibre (and hence of the intensity of light transmitted) varies according to the mechanical deformation of the fibre caused by pressure. Micro-bend sensors are typically used to measure the small pressure changes generated in vortex shedding flow meters, where a fibre-optic cable is stretched across the pipe.

4.3 Temperature

4.3.1 Thermocouple

Thermocouples are based on the principle that a temperature-dependent voltage is produced when dissimilar metals are connected together. The five standard base metal thermocouples are chromel-constantan (Type E), iron-constantan (Type J), chromel-alumel (Type K), nicrosil-nisil (Type N) and copper-constantan (Type T). These are all relatively cheap to manufacture, but become inaccurate over time and have a short lifespan (Chapter 14 in [17]). This drift in calibration over time is less of a problem in experiments than in actual reactors, where thermocouples are expected to provide readings for decades. Some types are sensitive to the operating environment and they can measure up to 1200°C with a typical inaccuracy of ±0.5% of full scale.

It is often impractical to connect a voltage-measuring instrument in close proximity to the point at which temperature is being measured, and so extension leads up to several meters long are required. This modifies the electrical circuit, and so care must be taken regarding the materials used in the extension leads and calibration of the thermocouple. If possible, the tip should be left exposed to maximise the speed of response, although they are delicate devices that must be treated carefully. Thermocouples can be protected using an insulated sheath (often called a probe), although this increases the time taken for the thermocouple to respond (e.g. ~0.15s time constant for 1mm diameter sheath).

Thermocouples respond quickly to changes in temperature, and are a simple, cheap and widely used method for measuring fluid and solid surface temperature.

4.3.2 Resistance Thermometer

Resistance thermometers or Resistance Temperature Devices (RTD) are based on the principle that the resistance of a metal varies with temperature. Platinum has the most linear resistance/temperature characteristic and it also has good chemical inertness (Chapter 14 in [17]). In the case of non-corrosive and non-conducting environments, RTDs are used without protection. In all other applications, they are protected inside a sheath, which reduces the speed of response to changes in temperature.

RTDs are commonly used up to 650°C with a measurement inaccuracy of ±0.5%. They are generally more stable than thermocouples and can measure small temperature differences.

4.3.3 Distributed Temperature Sensor

Distributed Temperature Sensors (DTS) use fibre optic cables to measure the temperature profile along the sensor cable because changes in temperature along the length of the cable alter the amount of light that is transmitted. Analysis of the backscattered radiation then enables temperature versus distance to be determined (Chapter 14 in [17]). High resolution DTS have a high measurement accuracy, and have been used to measure temperature distributions near the core of a research reactor and in corrosive liquid-sodium environments [19].

4.3.4 Thermography

Thermography, or thermal imaging, involves scanning an infrared (IR) radiation detector across an object. This is a non-invasive technique used to provide temperature measurements at a single point or on a surface area [20]. The radiation detector uses the principle that all objects emit electromagnetic radiation as a function of temperature. The temperature at the point that the instrument is focussed on is inferred from a measurement of the incoming infrared radiation.

Measurement resolution is high, with temperature differences as small as 0.1°C being detectable (Chapter 14 in [17]). However, the radiation from a body is very sensitive to the composition and surface condition of the body, i.e. its emissivity. Therefore, all radiation detectors have to be carefully calibrated for each particular body or by applying patches of known emissivity. In addition, IR thermography requires line of sight with the emitting body and the results can be affected by absorption and scattering between the emitting body and the detector e.g. dust and water droplets.

IR thermography has high sensitivity and fast response time, and can therefore be exploited to effectively measure convective heat fluxes with steady or transient techniques, or to perform detailed thermal surface flow visualisation [21]. IR thermography can also be used in situations where contact measurements would be hazardous or impossible, e.g. at temperatures exceeding 1300ºC [20]. However, the method is relatively expensive and can only be used to measure the temperature on a solid surface with gas present between the surface and infrared detector.

4.3.5 Temperature Sensitive Paint

Temperature Sensitive Paint (TSP) is a method for measuring surface temperature, usually in aerodynamic settings [22]. TSP is a paint-like coating which fluoresces under a specific illumination wavelength in differing intensities depending on the surface temperature due to thermal quenching that reduces the luminescent intensity. TSP is able to provide non-contact, high resolution, quantitative mapping of surface temperature on complex surfaces at low cost. A generic TSP measurement system is composed of paint, illumination light, photodetector and data acquisition/processing unit, with each pixel of the imaging camera acting as a thermocouple.

Time-resolved TSP applications involve pulsed excitation, and so depends on the ability to effectively detect the emission from TSP in short exposures. One can thus determine surface temperature variation as a function of time. In this case, the imaging devices must be synchronized to the excitation with multi-channel digital delay/pulse generators providing that synchronization. This technique is used in both aerodynamic and turbomachinery applications, and can be used to identify the transition from laminar to turbulent flow due to the abrupt change from low to high heat transfer.

Since TSP uses imaging cameras, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.4 Velocity

4.4.1 Pitot Tube

A pitot tube consists of two openings; one perpendicular to the flow, sampling local static pressure and one normal to the flow, sampling local total pressure (where the kinetic energy of the flow is converted to a pressure increase). The difference in measured static and total pressure is used to calculate the local fluid flow velocity (Chapter 16 in [17]).

Pitot tubes are a simple, cheap and widely used method for measuring velocity; however they have a low measurement accuracy. In addition, a pitot tube provides a single one-dimensional velocity, although pitot cylinders and pitot spheres can measure two-dimensional and three-dimensional velocities respectively.

4.4.2 Constant Temperature Anemometry

Constant Temperature Anemometry (CTA), also known as Thermal Anemometry, is a technique for the measurement of 1, 2 or 3-dimensional velocity and turbulence in gas and liquid flows, using hot-wire or hot-film probes inserted in the flow. CTA is particularly suitable for the measurement of flows with very fast fluctuations at a point (high turbulence) and the study of flow micro structures, where there is a need to resolve small flow eddies down to the order of tenths of a millimetre.

Hot-wire anemometers consist of a thin electrically heated wire and the measurement principle is based on the cooling effect of a flow on a heated body, which reduces its resistance. Measurement of this resistance change is used to calculate the fluid velocity. Hot-wire anemometers have a fast response time; however they are not very robust due to the small diameter of the wire (Chapter 16 in [17]).

Application areas include temperature, shear stress, velocity and turbulence measurements in e.g. jets, boundary layers and transitional flows. Wire sensors are used in gases and non-conducting liquids, while film sensors are primarily designed for use in water and other conducting liquids. However, the temperature range for CTAs is limited, and they require careful calibration.

The Thermal Anemometry Grid Sensor is based on the same principle; however it contains multiple temperature resistant resistors arranged in a grid. This enables spatially resolved velocity measurements to be obtained.

4.4.3 Wire Mesh Sensor

Wire mesh sensors (WMS) are an intrusive technique that allow the investigation of multiphase flows with high spatial and temporal resolution. They can be used to obtain information about fluid velocity, void fraction, droplet / bubble size and distribution, interfacial area, film thickness, thermal distribution and flow regimes [23].

Typically, a WMS consists of two parallel planes of wires; one plane containing transmitter wires, and one containing receiver wires. The planes are configured such that the wires on the top and bottom planes cross at an angle of 90°, forming a mesh grid of electrodes. This is placed into the cross-sectional area of the pipe or flow region of interest. The transmitter wires are activated sequentially, while the receiver wires are sampled in parallel. An electrical property (conductivity or permittivity) at each crossing point is evaluated to determine the fluid distribution across the cross-section.

Velocity measurements are typically obtained using two WMS in different locations, and high accuracy three-dimensional measurements can be made using two sensors separated by a few centimetres [23].

WMS can be used at typical reactor temperatures and pressures, but is intrusive and so can influence the flow field and size/shape of the bubbles. In addition, two phases are required in order to evaluate the conductivity/permittivity variation at each crossing e.g. air-water, steam-water or water with varying dissolved solute concentrations. This limits its applicability to velocity measurements, and therefore it is often used for measurements of mixing and void fraction (Section 4.6.5)

4.4.4 Particle Image Velocimetry

Particle Image Velocimetry (PIV) is a non-intrusive optical measurement technique which is used to obtain 2D or 3D fluid velocity measurements. PIV involves introducing small light scattering particles into the fluid flow. These particles are then illuminated by two consecutive pulses of a high intensity laser, and the scattered light across a plane of flow is recorded by a high resolution camera. The fluid velocity is then determined by analysis of the displacement of the particles between the successive frames.

Stereo PIV measures three velocity components in a plane using two cameras, with the second camera at a different orientation such that it can record the third dimension component. Tomographic PIV uses multiple cameras and can measure three velocity components in a volume. Time resolved PIV benefits from the advances in camera technology to acquire high resolution PIV images at tens of thousands of frames per second with full camera resolution.

In air flows, the seeding particles are typically oil drops in the range 1 µm to 5 µm. For water applications, the seeding is typically polystyrene, polyamide or hollow glass spheres in the range 5 µm to 100 µm. Since PIV uses scattered light, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

PIV can be applied to flow regimes ranging from laminar, transitional to fully turbulent flows [24]. PIV is typically used in single phase fluids, but can be applied to two-phase flow with some modification [25].

4.4.5 Laser Doppler Velocimetry/Anemometry

Laser Doppler Velocimetry (LDV) or Laser Doppler Anemometry (LDA) is a non-intrusive, optical technique for 1D, 2D and 3D point measurement of velocity and turbulence distribution in both free flows and internal flows. LDV requires that light scattering particles are present within the flowing fluid. The flow is then illuminated by a known frequency of laser, and the light scattered by the moving particles is detected by a photomultiplier. The difference in frequency between the incident light and light reflected back is measured (i.e. the Doppler frequency), and used to calculate the velocity.

LDV can be used in hostile environments and confined areas and can measure a wide range of velocities (Chapter 16 in [17]) with high spatial and temporal resolution. However, the method tends to be expensive and it can be difficult to collect data in close proximity to walls. Doppler Global Velocimetry is based on the same principles as LDV, but allows measurement of the velocity distribution within a plane.

LDV is used for single phase flow measurements; in two-phase flow, the beam is scattered by the interface between fluids, resulting in significantly reduced efficiency [25]. Since LDV uses lasers, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.4.6 Ultrasound Doppler Velocimetry

Ultrasound Doppler Velocimetry (UDV) is based on the pulse-echo method with velocity derived from shifts in positions between pulses. This requires acoustic inhomogeneities in the fluid to measure, which may be of natural origin as for many metal melts or artificial scattering particles have to be added. This technique is especially useful for opaque fluids or systems without optical access, since established optical flow measurement techniques such as PIV and LDV cannot be used [26].

2D or 3D Ultrasonic Doppler Velocimetry can be used to measure two or three velocity components simultaneously along a line. UDV-2D is based on a 3 transducers system with one transducer as an emitter and two others as receivers. For UDV-3D measurements, a similar arrangement is used with three receivers.

In addition, spatial flow measurements can be achieved using several transducers arranged laterally to each other. However, the spatial and temporal resolution is limited by the number of transducers and sequential excitation required.

4.5 Flow Visualisation

4.5.1 Tomography

Tomography encompass a number of techniques that allow visualisation of a two-dimensional image of a cross-sectional area of the measured object. Such techniques require the physical property to be measured at multiple locations simultaneously on the measurement plane. Reconstruction algorithms are then applied to produce a 2-D image, and various flow properties such as velocity and void faction can then be evaluated by analysis of successive frames.

Tomographic methods must have a high temporal resolution to detect the evolution of transient flows, and a sufficiently high spatial resolution such that they can detect the evolution of the flow structure [25]. Some tomographic methods include x-ray and gamma ray computed tomography, electrical conductance tomography, acoustic tomography and tomographic PIV. It is worth noting that x-ray and gamma ray equipment require special permits and trained personnel to operate with associated safety inspections, strict control of equipment and procedures that could impact other activities in the facility.

4.5.2 Radiography

Radiography involves obtaining a 2-D shadow image of the object being measured. This is achieved by focussing a beam of radiation (neutrons or photons) through the fluid. The beam attenuates through the fluid and is detected by a two-dimensional imaging device. The image contains information about the internal structure of the fluid as the beam is attenuated according to the basic exponential law of radiation attenuation [25].

X-rays, gamma rays and neutrons can be used as radio sources. These sources interact differently with matter, and so each source provides a different type of information, and each source can be applied more efficiently to certain environments. For example, neutron radiography is particularly useful as it can distinguish between different isotopes, and is efficient for visualising two-phase flows at high temperatures and pressures contained within thick walled vessels [25]. However, X-ray radiography is safer as the energy can be controlled by altering the input voltage. As stated in Section 4.5.1, radiography equipment require special permits and trained personnel.

4.5.3 Planar Laser Induced Fluorescence

Planar Laser Induced Fluorescence (PLIF) is a non-intrusive imaging technique used to provide 2D information on the magnitude of scalars: temperature, pressure or concentration [27]. It is a derivation of Laser Induced Fluorescence, which provides single point measurements. The equipment used in PLIF is similar to that used in PIV, with the key difference being the use of different seeding particles and a different optical path.

PLIF involves introducing a fluorescent dye into the fluid and illuminating the fluid with a thin laser light sheet. The fluorescent dye is excited by the laser and the light is captured by a high speed camera. PLIF can be combined with PIV to provide concentration/velocity or temperature/velocity measurements.

This information can be used to study heat transfer or mixing at the scalar transport level. PLIF has also been used in many other applications, including quantification of film thickness over a 2D domain [28]. Since PLIF uses lasers, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.5.4 High Speed Imaging

High speed imaging is a commonly used technique for flow visualisation. Time resolved visualisation is crucial for the investigation of transient flows. From a single experiment, features characterising the development of the flow can be identified, and image sequences can be used to give a qualitative description of the flow and a quantitative evaluation of the propagation of flow features [29].

A limitation of the time resolved imaging however, is that at high frame rates, the spatial resolution is compromised. For example, at frame rates above 100,000 fps, most cameras typically operate at a resolution of 0.1 megapixel or less [29]. Observing detailed elements within a flow field therefore remains a key challenge. In addition, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.6 Void Fraction

4.6.1 Tomography

X-ray Computed Tomography (CT) is a non-intrusive technique used to obtain two-dimensional tomographic images. The principle of tomographic techniques is described in Section 4.5.1. X-ray CT is commonly used in the medical industry, and has recently been applied to visualise the internal behaviour of two-phase flow [25].

X-ray equipment is often expensive and bulky; however x-rays have the ability to penetrate dense construction materials, and the images have high temporal and spatial resolutions [30]. Gamma ray tomography is based on the same principles as x-ray tomography. This method can measure stationary phase distributions with high accuracy, and as gamma-rays can penetrate through thick metal structures, this method is useful for larger test facilities and pressure vessels [31]. However, the technique is quite slow, and can only provide time-averaged phase distributions [31]. As stated in Section 4.5.1, x-ray and gamma ray equipment require special permits and trained personnel.

4.6.2 Radiation Attenuation

Radiation attenuation methods are based upon measuring the attenuation of radiation through the test specimen. The attenuation of radiation is greater in liquids and solids than in gas, and so the measured change in intensity can be used to determine the structure of the fluid.

The apparatus consists of a shielded source of radiation (x-rays, gamma rays or neutrons), with a collimator to focus a beam of radiation. A scintillator coupled to a photodetector is typically used to detect the radiation, where the scintillator absorbs the radiation and subsequently emits a flash of light.

A beam is used to give one-dimensional measurements, however a linear source can be used to obtain cross-sectional average void fraction measurements [25].

4.6.3 Electrical Methods

Electrical impedance methods are based on measuring the degree at which a circuit resists a current when a voltage is applied. The impedance is measured using electrodes, which can be arranged in various configurations. The electrical impedance of a two-phase fluid depends on both the void fraction and flow structure, and so measuring the void fraction requires knowledge of the flow pattern [25].

Obtaining the proper calibration curve is the main problem when employing this technique [25]. Various types of electrical impedance methods exist, including electrical resistance probes and electric capacitance sensors. Resistance probes are intrusive whereas capacitance sensors are non-intrusive.

4.6.4 Optical Probes

An optical probe is an intrusive device used to measure the local void fraction in gas-liquid two-phase flow. It consists of a glass fibre optic cable with a small diameter (100 µm) connected at one end to an infrared source and photo-diode [25]. The free end of the probe is inserted into the fluid being measured. If the free end is in air, then light emitted from the source will be completely reflected back. However, if the free end is in water, then the reflection coefficient will be approximately zero.

Reflected light is converted into an electric signal by the photo-diode, and this is used to determine the void fraction. Single tip optical fibre probes are used to measure instantaneous and time-averaged local void fractions, and recent developments have used multi-tip optical probes for measuring bubble size, velocity and flow regime transition [32].

4.6.5 Wire Mesh Sensor

The principle of WMS is described in Section 4.4.3. Although this is an intrusive method, it has many advantages for void fraction measurements including high sensitivity and no requirement for a reconstruction algorithm. In addition, it can produce instantaneous data at a rate of 1,200 frames per second with a high spatial resolution of approximately 2-3 mm [32].

4.6.6 Hot-Wire Anemometry

The principle of hot-wire anemometry for measuring velocity is described in Section 4.4.2; however hot-wire anemometers and hot film sensors are also used for measuring void fraction. When bubbles in the liquid enclose the wire or sensor, the voltage output indicates a sharp dip. The fraction of time that the bubbles enclose the wire is equivalent to the local void fraction

4.7 Film Thickness

4.7.1 Acoustic Methods

Acoustic methods are based on the principle that ultrasonic waves are attenuated and reflected when crossing discontinuities in a medium [33]. The transit time can therefore be used to measure the film thickness.

High frequency ultrasonic waves are advantageous as they provide improved signal to noise ratio and increased resolution. Acoustic methods are non-intrusive and can be used to measure ultra-thin films; film thickness measurements in the range of 50-500 µmm have been reported [33].

4.7.2 Electrical Methods

Conductance methods are based on imposing an electrical potential difference between electrodes and measuring the subsequent current [33]. The conductivity between the probes is proportional to the amount of fluid between them; a thicker film will have a higher conductivity than a thin film. Conductive methods are intrusive and have been used in some studies to measure film thicknesses in the range 50 µm to 1.2 mm [33].

Capacitance based methods are also used for film thickness measurements. Such methods are non-intrusive and are based on the principle that when two metal plates face each other, a capacitance is created. This is measured and correlated with film thickness. Film thicknesses in the range 0.4mm to 23mm have been reported in some studies using this method [33].

4.7.3 Optical Methods

Interface detection methods are based on detecting the interfaces between solid-liquid and liquid-gas. Separate phases are distinguishable as the light emitted from each presents different characteristics in terms of colour or intensity, with light gradients forming at the interface [33].

High speed cameras and photographs are used to obtain the image of the flow, and film thickness can be determined by a process of pixel counting. Induced fluorescent dye, seeding particles and coloured liquids are some of the methods used to create the light gradient at the interfaces.

Many other optical methods have been used for the measurement of film thickness. This includes shadow graphic methods, and methods based on light attenuation, total internal reflection of light, beam laser scattering, fluorescence intensity, beam laser scattering, laser focus displacement and interferometry [33]. As for other flow visualisation techniques (Section 4.5), the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.7.4 Radiation Attenuation

The principle of radiation attenuation methods are described in Section 4.6.2. These methods provide line averaged liquid fraction and not local film thickness measurements [33].

4.8 Vibration

4.8.1 Laser Doppler Vibrometry

Laser Doppler Vibrometry is a non-intrusive laser based optical technique used to measure the vibration of a surface.

It is based on the same principle as Laser Doppler Velocimetry (Section 4.4.5), where the Doppler shift of the laser is used to determine the frequency and amplitude of the vibration. For accurate measurements, the surface of the test specimen must be sufficiently reflective. Reflective tape has been used in a reactor core mock-up where the smooth steel surface of a fuel pin reflected an insufficient amount of light [34]. However, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.8.2 Grid Method

The grid method involves capturing 2D or 3D images of the test specimen and analysing the phase differences between grid lines in the images [34]. Some methods used to introduce the grid onto the test specimen include etching or physically attaching the grid onto the surface.

The grid method is non-intrusive and so does not introduce any additional mass to the test object. However, the grid has been shown to introduce a notable difference to the stiffness of the test specimen [34]. In addition, the test fluid and test section walls must be optically transparent, which makes it difficult to use at reactor pressures and temperatures.

4.8.3 Contact Sensors

A number of intrusive techniques for measuring vibration are also used. Some methods include: Fibre Bragg Gratings (FBG), strain gauges and accelerometers.

- The FBG is formed by a periodic modulation of the refractive index in the optical fibre core. The resonance wavelength at which the FBG reflects broadband light is a function of the axial strain applied [34], which can be used to obtain vibration measurements.
- A strain gauge is an electrical circuit that has a strain dependent resistance. The change in resistance can be measured using a Wheatstone bridge and then amplified.
- An accelerometer is a device that detects the acceleration of a test mass and converts it to an electrical signal [34].Types of accelerometers include piezoelectric accelerometers and Micro-Electro Mechanical Systems (MEMS) accelerometers.

5 Water Facilities

Water Cooled Reactors (WCRs) have played a major role in the commercial nuclear industry since its beginnings and presently account for more than 95% of all operating civilian power reactors worldwide. Furthermore, the majority of nuclear reactors under development and construction are water-cooled[5] [35].

Heavy Water Reactors (HWRs) use "deuterated" water, the molecules of which comprise hydrogen atoms that are made up to more than 99% of deuterium, a heavier hydrogen isotope. This heavy water, used as a moderator as well as the coolant, improves the overall neutron economy, allowing fuel to be used that does not require enrichment.

Light Water Reactors (LWRs) are types of thermal-neutron reactor that utilises normal water, as opposed to heavy water, as both its coolant and neutron moderator. LWRs are the most common type of WCR deployed worldwide and are grouped into three types:

- Pressurised Water Reactors (PWRs), which produce steam for a turbine in a secondary loop via separate steam generators;
- Boiling Water Reactors (BWRs), which use the steam produced inside the primary circuit of the reactor core directly in a steam turbine; and
- SuperCritical Water-cooled Reactor (SCWR), a light water Gen IV design, which operates at high temperature and high pressure above the thermodynamic critical point of water (374°C, 22.1 MPa).

The family of nuclear reactors known as LWRs, cooled and moderated using ordinary water, tend to be simpler and cheaper to build than other types of nuclear reactor. Due to these factors they make up the vast majority of civil nuclear reactors and naval propulsion reactors in service throughout the world.

As a result, a significant amount of research and development has been undertaken in this area, and numerous experimental test rigs have been developed to investigate nuclear thermal hydraulic phenomena in water reactors, validating computational tools and performing scaling analyses for light water reactors.

The OECD has recently developed The International Experimental Thermal Hydraulics Systems (TIETHYS) database[6]. This is an online database which allows users to identify and search for thermal hydraulic experimental data relevant for validation of numerical models and computer codes. In addition, the NEA Research and Test Facilities DataBase (RTFDB)[7] contains a searchable database, which includes a large number of thermal hydraulic test facilities.

The separate effect test facilities and integral effect test facilities for water reactors are discussed in Section 5.1 and Section 5.2 respectively, while the test facilities associated with the range of containment (ex-vessel) phenomena in water reactors are described in Section 5.3.

In addition, the following laboratories, research institutes and organisations that specialise in water test facilities have been reviewed in more detail:

- Stern Laboratory in Canada (Section 5.4)
- Framatone in Germany (Section 5.5)
- HZDR in Germany (Section 5.6)

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[5] https://www.iaea.org/topics/water-cooled-reactors
[6] https://www.oecd-nea.org/tiethysweb
[7] https://www.oecd-nea.org/rtfdb/public/search

- BARC in India (Section 5.7)
- SIET in Italy (Section 5.8)
- Hitachi in Japan (Section 5.9)
- TU Delft in Netherlands (Section 5.10)
- PSI in Switzerland (Section 5.11)
- Westinghouse in Sweden and US (Section 5.12)
- Texas A&M University in US (Section 5.13)

The current research focus and areas of expertise are highlighted for each facility, together with their instrumentation and measurement capability and a description of the operational test rigs. The main parameters for each test rig are included in Annex A.

5.1 Separate Effect Test Facilities

An internationally agreed SET Validation Matrix for thermal hydraulic system codes was developed in NEA/CSNI/R(1993)14 Part 1 and 2 in 1993 [36]. This matrix collected together the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties in the modelling of individual phenomena. In addition, it was intended to record information that was generated around the world, so that it is more accessible.

The methodology that was developed during the process of establishing the SET validation matrix is detailed in NEA/CSNI/R(1993)14 Part 1 and 2 [36], together with the test facilities selected for the SET validation matrix ordered by country. The test facilities in the SET validation matrix are also listed in the TIETHYS database[8].

This included the identification and characterisation of the thermal hydraulic phenomena relevant to two-phase flow related to LOCAs and thermal hydraulic transients in LWRs. The SET matrix was developed by selecting individual tests from the test facilities relevant to each thermal hydraulic phenomenon so that it could be used to assess transient thermal hydraulic system codes.

A summary of the separate effect test facilities for water reactors that have been built and operated after the original database was developed is listed in Table 3 in Annex A. The information on each facility has been taken from references, conferences and publications, and are ordered by country, and includes: organisation, maximum power, status, type of test and operating temperature and pressure.

5.2 Integral Effect Test Facilities

An internationally agreed IET matrix was developed in NEA/CSNI/R(1996)17 [37] in 1996 for the validation of best estimate thermal hydraulic codes. This identified the main physical phenomena that occur during the considered accidents and the tests suitable for replicating them were selected. The relevant experimental facilities were identified and a list of experiments was selected for each facility.

This matrix was intended to collect together the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties. Each matrix is composed of three sub-matrices, related to the following items:

- Phenomena versus test types;

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[8] https://www.oecd-nea.org/tiethysweb

- Phenomena versus test facilities; and
- Test facilities versus test types.

Integral test facilities aim to represent the main features of the reactor coolant system including the main safety systems. Therefore, as part of the matrix development, accident scenarios were identified and classified. These classifications are slightly different for PWRs and BWRs (NEA/CSNI/R(1996)17) [37], and Water-Water Energetic Reactors (VVERs) (NEA/CSNI/R(2001)4) [38]. All of the test facilities within the integral test facility matrix are also listed and described in the TIETHYS database[9].

A summary of the integral effect test facilities for water reactors that have been built and operated after the original database was developed is listed in Table 4 in Annex A. The information on each facility has been taken from the TIETHYS database, published papers, conferences and publications. They are ordered by country, and include: organisation, maximum power, reference reactor, status, height scale, volumetric scale and operating pressure.

5.3 Containment Test Facilities

The CSNI formed the Containment Code Validation Matrix task group in 2002. The objective of this group was to define a basic set of available experiments for code validation, covering the range of containment (ex-vessel) phenomena expected during light and heavy water reactor Design Basis Accidents (DBA), Beyond Design Basis Accident (BDBA) and Severe Accidents (SA). This is intended to complement the SET [36] and IET [37] CSNI validation matrices for thermal hydraulic code validation.

The Containment Code Validation Matrix NEA/CSNI/R(2014)3 [39] provides a brief overview of the main features of a PWR, BWR, Canada Deuterium Uranium (CANDU) and VVER reactors. The main focus is to identify the phenomena and safety systems employed in these reactor types and to highlight the differences.

The Containment Code Validation Matrix [39] contains a description of 127 phenomena, broken down into 6 categories. The two categories most relevant to thermal hydraulics are containment thermal hydraulics and hydrogen behaviour (Combustion, Mitigation and Generation) phenomena.

A summary of the main test facilities for containment thermal hydraulics and hydrogen behaviour (combustion, mitigation and generation) phenomena is given in Table 5 in Annex A. The information on each facility has been taken from the Containment Code Validation Matrix [39], as well as references, conferences and publications, and are ordered by country, and includes: organisation, status, type of test and operating temperature and pressure.

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[9] https://www.oecd-nea.org/tiethysweb

5.4 Stern Laboratories Inc in Canada

Stern Laboratories Incorporated[10] is an employee-owned private Canadian company located in Hamilton, Ontario, in the former Westinghouse Canada manufacturing plant. It was formed in 1988 and has been operating for over 30 years in the nuclear industry worldwide [40].

Stern is an approved supplier for many utilities, reactor and fuel designers, government agencies and other laboratories, and provides the following services:

- Heat transfer experiments for CANDU, BWR and PWR applications.
- Design and fabrication of heated fuel simulators.
- Design and manufacture of specialised equipment.
- Computational Fluid Dynamics of reactor thermal hydraulics.

Stern's quality assurance program satisfies the requirements for 10CFR50 Appendix B, NQA-1, 10CFR part 21 and applicable elements of the Canadian Standards Association (CSA): CSA N286-05, CSA Z299.2 and CSA N286.2 Clause 6. This includes performing qualification testing in accordance with CSA N286 Clause 6.

Stern Laboratories occupies an area of 3,700 m^2, employs about 35 people and has an on-site fabrication shop and clean room. The overall capability of the Stern site is listed in Table 1.

Total DC Power	16.75 MW
Maximum system pressure	18 MPa
Maximum flow rate	45 kg/s
Maximum temperature	340°C

Table 1: Overall Stern Capability

There are 13 independently controlled DC power zones: 5x2.5 MW, 3x1 MW and 5x0.25 MW supplies with accurate power metering of each power supply and fuel simulator. The oxygen content, conductivity and pH level of loop water are monitored during operation and hydrazine can be injected to minimize the dissolved oxygen content.

Stern Laboratories specialise in critical power tests (Section 5.4.1), but also have a number of other facilities available to perform reactor safety and reliability experiments, including:

- High pressure and temperature loop up to 11.5 MPa and 315°C that can investigate acoustic pulsations in a horizontal CANDU fuel channel.
- Low pressure and temperature loop up to 1.5 MPa and 150°C to undertake pressure drop, flow induced vibration and endurance testing of fuel assemblies.
- Single phase loop up to 11 MPa and 315°C for pressure drop and endurance testing of BWR fuel components.
- Vertical channel to undertake counter-current flow limiting experiments.
- Corrosion test loop up to 11 MPa and 320°C for testing samples under controlled conditions.
- High pressure and temperature loop up to 18 MPa and 371°C with nitrogen injection to investigate the impact of dissolved nitrogen on CHF at reactor conditions.
- Full-scale CANDU pressure tube burst facility to investigate tube burst at reactor conditions.

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[10] http://sternlab.com/

5.4.1 Critical Power Tests

Critical Heat Flux (CHF) - The full scale, horizontal CANDU fuel channel facility can perform CHF experiments on fuel elements that are up to 6 m long. This can accommodate complete fuel strings using directly heated fuel simulators. Non-uniform axial and radial power profiles can be applied over the full range of reactor conditions with pressures up to 13 MPa [41].

Boiling Transition (BT) - The full scale, vertical BWR loop facility is used for BT experiments with heated lengths up to 3.81 m using indirectly heated fuel simulators and up to 12 internal thermocouples per rod. This can accommodate up to 10x10 full-scale assemblies with grid spacers over the full range of reactor conditions with pressures up to 10 MPa.

Departure from Nucleate Boiling (DNB) - The vertical PWR loop facility is used to perform DNB experiments using indirectly heated fuel simulators with up to 10 internal thermocouples per rod. This can accommodate up to 6x6 fuel assemblies with heated lengths up to 4.27 m over the full range of reactor conditions with pressures up to 18 MPa.

Uniform or non-uniform (cosine, inlet-peaked, or outlet-peaked) axial power profiles can be applied and each fuel simulator can be powered individually. The fuel simulators are equipped with moveable internal thermocouples i.e. the internal thermocouples can slide and rotate to monitor temperatures over most of the fuel string surface.

5.5 Framatome in Germany

Framatome[11], previously a subsidiary of AREVA NP, operate thermal hydraulic test facilities from three technical centres at Erlangen and Karlstein in Germany and Le Creusot in France with over 40 years' experience. These sites operate full-scale and scaled models of systems and components [42] [43]. This includes:

- Qualification of pumps and valves and auxiliary system components.
- Integral system tests for PWRs and BWRs.
- Heat transfer experiments focused on fuel assemblies and heat exchangers.
- Flow mixing and FIV for fuel assemblies and core components.

Framatome's quality assurance program satisfies the requirements for ASME NQA-1, is accredited as a test and inspection body according to ISO 17025 and ISO 17020, and is accepted by ILAC.

The thermal hydraulic test facilities occupy an area of over 2,000 m^2 with heights up to 32 m, and employs about 100 engineers and technicians. Framatome operate a range of thermal hydraulic test facilities at Karlstein, Erlangen and Le Creusot, which are described in Section 5.5.1, Section 5.5.2 and Section 5.5.3 respectively.

5.5.1 Karlstein Technical Centre

- The KATHY loop (KArlstein Thermal HYdraulic test loop) performs CHF tests on full-scale PWR and BWR fuel assemblies with electrically heated rod bundles. The KATHY loop can apply a range of axial power profiles (cosine, top-peak, down-peak and uniform) to test bundle geometries up to 12x12. The thermocouples inside the heated rods enable the axial and radial onset of CHF inside the test bundle to be determined. In addition, void fraction is measured using gamma ray 2D computed tomography and densitometer.
- The BENSON facility is a high pressure separate effects test facility that can investigate single and two-phase flows up to supercritical pressure conditions. It is a flexible facility that can be used to test heat transfer, pressure drop and natural circulation limits on a range of components e.g. condensers and steam generators.
- GAP is a large valve test facility, which is one of several valve test facilities on the site. This is used to qualify valves at reactor conditions for water, steam and two-phase flow conditions.
- APPEL (AREVA pump test loop) is used to qualify pumps to ISO 9906. This includes thermal transients, particle load and endurance tests.
- The INKA (INtegral Test Facility KArlstein) test facility was designed to simulate the performance of the passive safety systems of KERENA, the AREVA BWR design [44] with a volume scaling of 1:24. The RPV is simulated by the GAP accumulator vessel, which has a capacity of 125 m^3 at a pressure of 11 MPa. The large power supply at the facility (22 MW) means that INKA is capable of simulating various accident scenarios. Overall, there are over 300 sensors at INKA e.g. temperature, mass flow and pressure measurements. In addition, Thermo Needle Probes and Gamma densitometers are installed for two-phase flow instrumentation in cooperation with HZDR. The gas mixture in the vessels is measured via a probe sampling system analysed by a mass spectrometer in cooperation with PSI.

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[11] http://www.framatome.com/EN/customer-3832/thermalhydraulic-and-component-testing.html

5.5.2 Erlangen Technical Centre

- The KOPRA test facility is used to qualify and test full-scale PWR and BWR components at reactor conditions. It is a multi-functional test facility which includes four test loops for valves, core components, control rod drive mechanisms and fuel assemblies. KOPRA can also perform endurance tests to investigate long-term behaviour and wear.
- The PETER (PWR fuel Element Tests at ERlangen) test facility is used to investigate vibration and bowing in full scale PWR fuel assemblies. The vibration and movement of the fuel assemblies is measured using laser triangulation and laser vibrometers. In addition, the detailed flow velocity between the fuel rods is measured using LDV.
- The PKL (PrimärKreisLauf) integral test facility [45] has been used to investigate the thermal-hydraulic behaviour of PWRs under accident conditions for over 30 years. The PKL facility simulates the entire primary side and significant parts of the secondary side of a 1300 MW PWR at a height scale of 1:1. Volumes, power ratings and mass flows are scaled with a ratio of 1:145. The facility has been used to investigate a range of accident scenarios, including large, medium, and small breaks, shutdown procedures and complex thermal hydraulic phenomena with over 1500 measurement points. Since 2001, the PKL Project (PKL III E+F) was continued as part of an international NEA project. The subsequent PKL2 Project[12] (PKL III G), which ran from 2008 to 2011, investigated safety issues relevant to current PWR plants as well as new PWR design concepts.

5.5.3 Le Creusot Technical Centre

- The MAGALY test bench is used to measure flow-induced vibration in control rod guide tube assemblies. This is one of a number of FIV test facilities on the site. The movement and vibration is measured using laser vibrometers, strain gauges, accelerometers and displacement sensors.
- ROMÉO & JULIETTE are transparent scaled models of the RPV upper and lower plenum respectively. These models investigate and assess the flow distribution and pressure drop within the EPR RPV using PIV and LDA measurements of the velocity flow field.

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[12] https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-6.pdf

5.6 HZDR in Germany

Helmholtz-Zentrum Dresden-Rossendorf (HZDR) is located in Dresden, Germany and was founded in 1992, although research at the site dates back to 1955[13]. The experimental thermal fluid hydraulics department, part of the Institute of Fluid Dynamics, undertakes nuclear safety research and conducts experiments on multiphase flows with heat and mass transfer.

HZDR has developed high spatial and temporal resolution sensors and measurement techniques to support their multiphase experiments, including needle probes, tomography (X-ray, gamma and optical) and wire-mesh sensors, which provide high resolution data for CFD validation. The facilities at HZDR include:

- The TOPFLOW (TwO Phase FLOW) facility was commissioned in 2003 to study transient two-phase flow phenomena [46], [47]. TOPFLOW is a multipurpose facility with 3 test rigs: a pressure tank, vertical test loop and tomography laboratory.
    - The pressure tank is 6 m long with a diameter of 2.4 m and operate up to 5 MPa and 50°C. Different horizontal test sections e.g. hot leg, pressurized thermal shock and Direct Condensation and Entrainment Installation for Steam Experiments (DENISE) test rigs, can be installed inside the tank with large glass observation windows, as the pressure inside and outside the test section is at equilibrium. Measurements include up to 200 thermocouples and wire-mesh sensors, as well as high speed and infrared cameras.
    - The vertical test loop is used to study two-phase and counter current flows in vertical pipes (DN50 and DN200) up to 7 MPa with almost 8 m long test sections. The loop can investigate air-water and steam-water flows with the gas injected through modules at various points. The void fraction is measured using wire-mesh sensors on the DN200 pipe, and fast X-ray tomography on the DN50 pipe using the ROFEX (ROssendorf Fast Electron beam X-ray tomography) imaging technique developed by HZDR, which can take 8,000 images per second.
    - The tomography laboratory is a radiological shelter that is 7 m long x 3.4 m wide x 2.3 m high, which is connected to the TOPFLOW facility. This has been used to study steam condensation in a slightly inclined tube and boiling effects in a vertical test section using X-ray tomography.
- The TOPFLOW+ test facility was commissioned in 2014 to undertake smaller experiments under atmospheric conditions with 7 test areas over an area of 750 m^2, including a radiation-shielded X-ray laboratory. The facility has two indoor cranes and a small mechanical workshop, and includes water, air, oxygen, nitrogen and carbon dioxide supply. The X-ray laboratory has a ROFEX imaging system that is mounted on a traversing device, which can be transported around the facility, as well as a micro-focus X-ray tomography system that can resolve objects in the micrometer range. Two-phase flow measurements can also be taken using radioactive particle tracking, hot wire anemometry, wire mesh sensors and fast responding oxygen needle probes.
- The ROCOM (ROssendorf COolant Mixing model) facility is used to study mixing of coolant temperature and boron concentration in the RPV of a PWR using a salt tracer [48]. It is a 1:5 scale model of the KONVOI PWR that is constructed from acrylic glass. The transient salt concentration is measured using a wire-mesh sensor at 1,000 Hz. Wire-mesh sensors

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[13] https://www.hzdr.de/db/Cms?pNid=393&pOid=49734

are included in the coolant pipes (8.9 mm pitch), downcomer (13 mm radial and 5.625° aximuthal pitch) and core inlet (2mm pitch or 1 measurement per fuel assembly).

- HAWAC (Horizontal Air/WAter Channel) has an 8 m long rectangular, acrylic glass test section (100 mm wide and 30mm high) that is used to study co-current flows [49]. The air (upper) and water (lower) are injected separately and come into contact at the end of a 500 mm long blade. This blade can be inclined to control the cross-section of each phase. Optical measurements of the interface are made using a high-speed video.
- The GP Loop (Gas Particle Loop) is used to study the transport, deposition and resuspension of particles in turbulent flows e.g. graphite dust [50]. The 10 cm x 10 cm test section is made from acrylic glass and located 15 hydraulic diameters downstream of the particle injection point to ensure uniform particle distribution. The flow field was measured using PIV (planar and stereoscopic), and the deposited particles were counted using an optical and scanning electron microscope. In addition, multilayer deposition was studied using a laser distance sensor.

5.7 BARC in India

The Reactor Engineering Division of the Bhabha Atomic Research Centre (BARC)[14] undertake thermal hydraulics research and testing on Advanced Heavy Water Reactors (AHWR) at their main facility in Trombay, Mumbai. In addition, the AHWR Thermal-Hydraulic Test Facility (ATTF) has been running since 2013 at the R&D Centre, Tarapur to further validate AHWR design margins for CHF, parallel channel instability, operational transients and LOCA [51].

Imperial College London and the University of Leeds are currently working with BARC as part of a UK-India civil nuclear collaboration on thermal hydraulics for boiling and passive systems.

The AHWR being designed in India is a 920 MWth pressure tube type boiling light water cooled and heavy water moderated reactor. In the AHWR, it is proposed to remove the core heat by natural circulation during start-up, normal operation, transients and accidental conditions. The AHWR uses several passive concepts with a view to simplify the design and enhance safety.

Therefore, the Reactor Engineering Division of BARC do a significant amount of research and testing of natural circulation, flow instabilities and passive cooling systems using the following test facilities:

- The Boiling Water Loop has been used to undertake CHF, post dryout, mixing and single and two-phase pressure drop experiments [52]. This test facility can accommodate a full-length AHWR fuel channel.
- The HPNCL (High Pressure Natural Circulation Loop) is used to investigate two-phase natural circulation [53]. The HPNCL has a loop diameter of 50 mm, elevation of 3 m and heated section of 1 m. This loop has been used to study steady state natural circulation and stability behaviour, as well as the start-up procedure and parametric studies on the stability of natural circulation.
- The Apsara natural circulation loop is located in the Apsara reactor hall and uses neutron radiography to investigate flow pattern transition criteria, natural circulation instability and CHF [54] for a range of loop diameters (3/8”, 1/2”, 3/4” and 1”). The loop is about 2 m long and 2.5 m high with a 10 kW direct electrically heated test section. The main measurements include temperature, flow rate, pressure drop and void fraction. Void fraction is measured using neutron radiography and conductance probe methods.
- The PCL (Parallel Channel Loop) facility has four parallel channels, which are direct electrically heated sections [53]. The PCL facility is 3 m high with a pipe diameter of 25 mm, and has been used to investigate thermal hydraulic stability of parallel channels in natural circulation. The heater is 1 m long with a diameter of 12 mm and an operating pressure of 1.5 MPa. The loop is well instrumented and the channel flow is indirectly measured based on the single-phase pressure drop in a horizontal section.
- The FISBE (Facility for Integral System Behaviour Experiments) loop is a scaled representation of the AHWR Primary Heat Transport system. The volume scaling ratio is 1:76.5 based on a power to volume scaling philosophy with a 1:1 height scale. The instrumentation includes temperature, flow rate, pressure, density and level to investigate single phase natural circulation around the loop under different conditions, as well as LOCA and two-phase natural circulation.
- The ITL (Integral Test Loop) simulates the Main Heat Transport (MHT), Emergency Core Cooling System (ECCS), Isolation Condenser (IC) system and Gravity Driven Cooling

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[14] http://www.barc.gov.in/rddg/rddg/mod_rddgact/

System (GDCS) of the AHWR to investigate the overall system behaviour under different operating conditions, transients and accidents [55]. The scaling philosophy is based on a three level approach. The ITL has a volume scaling ratio of 1:452, height scale of 1:1, pressure (7 MPa) and temperature (285°C) scale of 1:1 with a maximum power to the test section of 3 MW [55]. The ITL has been used to investigate plant transients and LOCA scenarios, as well as natural circulation in the MHT loop including parallel channel and asymmetric behaviour.

- The ATTF (AHWR Thermal Hydraulic Test Facility) is an integral test facility that simulates the thermal hydraulic behaviour of the MHT system and ECCS that became operational in 2013 [51]. The scaling philosophy is based on a three level approach with the global scaling based on a power to volume ratio of 1:226. The ATTF has a maximum power of 9 MW for parallel channel stability tests, and is well instrumented with pressure, flow, level and temperature measurements at various points. The ATTF is used to investigate thermal margin and parallel channel stability behaviour.

5.8 SIET in Italy

The SIET Company[15] (Società Informazioni Esperienze Termoidrauliche) was set up in 1984 by ENEA and CISE with the aim of creating a reference point in Italy for thermal hydraulic experimental research in the field of NPP safety. It is located in Piacenza inside the 'Emilia' disused power station.

SIET is a testing and R&D centre that is able to test NPP components at realistic reactor conditions. Its core business is thermal hydraulic testing for R&D and certification of components and systems, including heat exchangers, valves, steam-water separators, steam injectors and heat removal systems.

SIET offices and facilities occupy an area of 5,000 m^2 with a height of 30 m and has an on-site calibration laboratory [56]. SIET employs about 23 people and the overall capability of the site is listed in Table 2.

Total AC power	9 MW
Total DC power	6.5 MW
High pressure water @ 17 MPa, 330°C	200 kg/s
Saturated steam @ 7 MPa 330°C	40 kg/s
Superheated steam @ 10 MPa, 500°C	8 kg/s

Table 2: SIET facility capability [56]

SIET holds a number of international accreditations, including ISO 9001, ISO 17025 and CEN (European Committee for Standardization) recognised 'Test House' and 'Empowered Certification Body', Accredia (Italian National Accreditation Body) and ILAC. SIET also has a nuclear quality assurance culture in place that conforms to 10 CFR 50 Appendix B, 10 CFR 21 and ASME NQA-1.

SIET operates a number of thermal hydraulic test facilities, which include:

- IETI (Impianto per Esperienze Termo-Idrauliche) is a multi-purpose facility for basic heat transfer, special devices and special instrumentation [56]. IETI tests different plant components using water and steam under high pressure, high temperature conditions. It can be used for a wide range of thermal hydraulic experiments under steady and transient conditions from basic experiments to full scale power channels with a maximum test section height of 15 m. This includes CHF tests, as well as R&D on steam jet pumps and steam injectors. In addition, the IETI facility has been used to undertake development tests on helical coil steam generator coils to investigate pressure drop, heat transfer coefficient, CHF and flow instability.
- GEST (Generatore Esperienze Sicurezza Termoidrauliche) facility for large scale components and heat removal systems. GEST-SEP is used to test both BWR and PWR steam generator separators and dryers at typical reactor conditions. This facility was started in 1984 and is mainly used for large scale performance and qualification tests of components working in the presence of a water-steam mixture. The GEST-SEP facility allows the operation and efficiency of steam-water separators to be tested within a 43 m^3 vessel with a maximum height of 20 m and maximum diameter of 3 m. The GEST-GEN facility was started in 1985 to support the development of PWR steam generators. Test

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[15] https://www.siet.it/home-page.html?lSite=en

facility applications include large scale thermal hydraulic tests for steam generators and qualification of steam-water heat exchangers.

- The PANTHERS (Performance ANalysis and Testing of HEat Removal Systems) test facility has performed functional and endurance tests on SBWR Isolation Condenser (IC) and Passive Containment Cooling (PCC) systems [57]. This facility began operating in 1994 and tested two types of condensing heat exchangers under typical operating conditions with sufficient measurements to assess their thermal hydraulic performance and mechanical integrity.
- The PERSEO (in-Pool Energy Removal System for Emergency Operation) test facility was used to test an innovative IC heat removal system by modifying the PANTHERS IC-PCC plant in 2002 [58]. The primary side of the facility consists of a pressure vessel (43 m^3 volume, 13 m height) and a full scale SBWR IC heat exchanger module. The pool side of the facility consists of the pool containing the heat exchanger (~30 m^3 volume), and the main pool representing the water reservoir (~200 m^3 volume). The pressure vessel is maintained at saturation using de-superheated steam supplied from the nearby Edipower power station.
- SPES (Simulatore Per Esperienze di Sicurezza - Simulator for Safety Tests) is a PWR integral test facility. The SPES integral test facility was first operated in 1988 for SPES-1, and was then converted to SPES-2 in 1992. SPES-2 is a 2-loop PWR integral test facility that was operated from 1992 to 1994, which contributed to the US NRC licensing of the AP-600 reactor. It includes the full-height primary circuit, secondary circuit, passive safety systems and non-safety systems [59] with a volumetric scaling ratio of 1:395. SPES-2 was used to undertake static and dynamic tests of accident and operational transients and investigate the performance of passive safety systems. SPES-3 has been designed to represent the IRIS reactor and partially built.

Other tests that are performed in the SIET facilities include:

- Heat exchanger performance tests i.e. heat transfer, efficiency, pressure drop and tube vibration.
- Pump performance tests i.e. head, power and efficiency.
- Piping and fitting tests, such as high pressure, high temperature tests, pressure drop, vibration and noise measurements.

5.9 Hitachi in Japan

Hitachi-GE Nuclear Energy Ltd[16] was established in 2007 as a joint venture between Hitachi and General Electric. Hitachi has over 50 years' experience in the design, manufacture and delivery of BWRs and the current Advanced Boiling Water Reactor (ABWR) design.

The UK ABWR is being developed by Hitachi-GE Nuclear Energy Ltd for the UK and has received design acceptance confirmation from the ONR in December 2017. Horizon Nuclear Power is planning to build UK ABWRs at Wylfa Newydd on the Isle of Anglesey and Oldbury-on-Severn in South Gloucestershire.

As part of the BWR design process, Hitachi have developed and operated a number of thermal hydraulic test facilities [60], which are focused on two-phase flow and verifying the performance of components:

- The HICOF (HItachi COre and Fuel thermal hydraulic test loop) was operated from 1984 to 1989 to run critical power tests on 1/4-sized fuel bundles and pressure drop tests on full-sized fuel bundles. This included transient boiling transition tests with chopped cosine axial power profiles and a radial power distribution within the fuel bundle.
- The FIVE (Flow Induced Vibration Experiment loop) is 10 m high with a 2 m test section and was operated from 2007 to 2008. Void fraction and tube acceleration were measured, as well as temperature, pressure and flow rate around the loop.
- The HUSTLE (Hitachi Utility Steam Test LEading) facility is 15 m high and was commissioned in 2009. This can test half-sized chimneys and fuel bundles, fuel rods, heat exchangers and pumps in steam, water and steam-water flow conditions. Measurement techniques include void fraction using wire mesh sensors and X-ray computed tomography, and film thickness on the rod surface using ultrasonic sensors.
- The STNS (Steam Test facility for Nuclear Steam system) is used to assess heat exchanger performance, and includes measurements of tube temperature and void fraction in the pool using a laser void sensor.

Hitachi-GE have also developed a 3D void fraction measurement system that uses X-ray computed tomography to provide improved resolution of two-phase flows.

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[16] http://www.hitachi-hgne.co.jp/en/about/index.html

5.10 TU Delft in the Netherlands

The Reactor Physics and Nuclear Materials group at the Delft University of Technology (TU Delft)[17] in the Netherlands undertake thermal hydraulics research using experimental facilities and numerical codes.

The Transport Phenomena and Nuclear Applications lab undertakes research in supercritical heat transfer using LDA and PIV techniques, as well as molten salts, isotope production and rod bundle flows. The test facilities at TU Delft include:

- The GENESIS facility was built in 2005 to study boiling natural circulation [61]. The facility is designed to represent the ESBWR design and uses Freon 134a as the scaling fluid. This significantly reduces the power required to match the ESBWR conditions to investigate flows, void fraction and density wave instabilities around the loop. The facility includes 25 heated rods, which can be run with constant power or simulated neutronic feedback. Measurements include temperature, void fraction, power and pressure drop.
- DeLight (Delft Light water reactor facility) was built in 2009 to investigate natural circulation stability in the High Performance Light Water Reactor (HPLWR), which is a SCWR design [62]. This facility is nearly 10 m high and uses Freon R23 as the scaling fluid. This was designed using scaling rules to represent the HPLWR pressure, temperature and power requirements, and hence simulate the stability behaviour. The rods are heated electrically with the ability to include neutronic feedback to investigate the impact on natural circulation stability. The loop includes pressure, temperature and flow rate measurements. The DeLight facility has also been modified as part of the THINS (ThermoHydraulics of Innovative Nuclear Systems) project to investigate supercritical turbulent flow using LDA measurements.
- The SEEDS (SEven rods bundle Experiments in Delft for SESAME)[18] facility was commissioned in 2017 to investigate fluid-structure interaction. Initial experiments were conducted using SEEDS 1 (hexagonal rod bundle), and the facility was converted to SEEDS 2 (wire wrapped bundle) in 2018. The bundle is made from refractive-index matching Fluorinated Ethylene Propylene to allow PIV and LDA measurements of the flow field inside the rod bundle. In addition, the central rod is flexible to allow it to vibrate with the movement captured using a high-speed camera. The results will provide benchmark test data for the SESAME (Simulations and Experiments for the Safety Assessment of MEtal cooled reactors) project.

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[17] https://www.tudelft.nl/en/faculty-of-applied-sciences/about-faculty/departments/radiation-science-technology/reactor-physics-and-nuclear-materials/

[18] http://sesame-h2020.eu/seeds-experimental-facility-revealed/

5.11 PSI in Switzerland

The Experimental Thermal-Hydraulics group[19] and Severe ACcident REsearch (SACRE) group[20] are part of the Laboratory for Reactor Physics and Thermal-Hydraulics at the Paul Scherrer Institut (PSI) in Switzerland.

5.11.1 Experimental Thermal-Hydraulics group

The experimental thermal-hydraulics group undertake experimental tests associated with nuclear containment, mixing processes and single and two-phase flow in rod bundles. The facility specialises in flow visualisation techniques, such as PIV, LIF, high speed cameras and infrared imaging, as well as mass spectroscopy and the development of fast neutron imaging techniques. Their facilities include:

- The LINX (Large scale Investigation of Natural circulation and miXing) facility is used to investigate mixing and condensation in the presence of non-condensable gases with a 9.42 m^3 pressure vessel [63]. The instrumentation includes multiple thermocouples and gas concentration sensors, as well as 12 glass windows for photography, PIV and infrared measurements. LINX is being used to study liquid film dynamics under condensation and re-evaporation conditions with near-infrared for the film thickness measurements and mid-infrared wavelength to measure the film surface temperature.
- The HOMER (HOrizontal Mixing Experiment in a Rectangular channel) test facility is used to study the mixing of different density gases (nitrogen and helium) under stable and unstable conditions in a horizontal mixing section [50]. The velocity of each stream is independently controlled by GAMILO (an open GAs MIxing LOop), which can modify the gas velocity and composition. Flow velocity and gas concentration is measured using PIV and LIF.
- The AROBE (Adiabatic ROd Bundle Experiments) facility is used to study two-phase flow in rod bundles under adiabatic air-water conditions. Liquid film thickness is measured by neutron imaging using the ICON facility[21].
- The CNHT facility is used to study convective boiling and dry out in a heated fuel bundle model with Chloroform as the scaling fluid to reduce the power required. Heavy water is used to heat the pins, and the facility can operate in sustained dry out. High resolution measurements are achieved using cold-neutron and X-ray imaging. PSI is experienced at cold-neutron imaging and has developed a fast neutron tomography technique (TwoFast) with fast neutrons and fast imaging detectors with a spatial resolution of 2 mm.
- SUBFLOW (SUB-channel FLOW test facility) contains an up-scaled (1:2.6) 4x4 fuel rod bundle model that is made of transparent material. An air injection system is used to ensure uniform bubble distribution and size in each sub-channel. This contains two 64x64 wire-mesh sensors 15mm apart to measure the single and two-phase flow field with high spatial (2 mm) and temporal (1250 Hz) resolution.
- FLORIS is 1:10 scale, two-dimensional (20 mm thick) vertical section of the downcomer, lower plenum, upper plenum and core to study density driven flow in the lower plenum of a BWR. The tracer concentrations throughout the domain are measured using a wire-mesh sensor with 10 mm spatial resolution.
- The GEMIX (GEneric MIXing Experiment) facility investigates mixing of two water streams downstream of a splitter plate with different densities, identical velocities and 4% free

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[19] https://www.psi.ch/teg/
[20] https://www.psi.ch/sacre/
[21] https://www.psi.ch/sinq/icon/

stream turbulence. The velocity field is measured using PIV, while the concentration field is measured using LIF and WMS. This facility has been used for the 4th OECD/NEA CFD benchmark with Uncertainty Quantification (turbulent mixing), which started in 2015, and has not yet been formally reported.

- The T-junction geometry is used for basic mixing studies using wire-mesh sensors downstream of the T-junction. This enables the transient mixing of the hot and cold streams to be measured.
- PANDA (Passive Nachwärmeabfuhr- und Druckabbau-Testanlage) is a large-scale test facility to study containment system behaviour [64]. It consists of 6 main vessels with a total volume of 515 m^3 and height of 25 m and has the ability to inject air, steam, helium and water into the vessels. Instrumentation includes flow meters, thermocouples and hot wire anemometers, as well as PIV and mass spectrometers. A test was conducted in 2013 for the OECD/NEA PSI CFD benchmark exercise (jet erosion of a stratified atmosphere) [65]. This benchmark investigated the erosion of a hydrogen-rich, stratified layer in top of a containment volume by the impingement of a buoyant, vertical jet.

5.11.2 Severe ACcident REsearch (SACRE) group

The severe accident research group conducts plant analyses using severe accident codes and analysis of aerosol transport and removal, as well as experimental studies into iodine retention, aerosol transport and two-phase flow hydrodynamics. Their facilities include:

- The DRAGON (DiveRse purpose Aerosol GeneratiON) facility can generate aerosols using a plasma torch system, fluidised bed and atomisation of suspended particles [66]. The facility includes aerosol and iodine generation, as well as the ability to mix the aerosols with steam, nitrogen or air. The aerosol composition can be measured using impactors, particle counters, photometers and filters. This has been used to generate test aerosols for other facilities e.g. VEFITA.
- The VEFITA (Venting Filter Assessment) facility is used to investigate filtered containment venting systems. This is a mock-up of a wet scrubber with reduced diameter and 1:1 height scale. Aerosols from the DRAGON facility are combined with steam/nitrogen to assess filter performance and determine decontamination factors. Instrumentation includes phase doppler particle analysers for droplet size and velocity and LDV for flow field velocity.
- ISOLDE investigates bubble hydrodynamics relevant to pool scrubbing applications, such as iodine mass transfer. The test section consists of 0.2 m diameter glass tubes with a total height of 5.5 m. Air, nitrogen or steam mixtures can be injected at variable heights and flow rates. A three-layer WMS with 3 mm spatial resolution is used to measure the bubble size and velocity. Instrumentation includes temperature, flow rate, pressure and water level.
- The TRISTAN (Tube Rupture In Steam generator multi-phase flow investigations) facility is used to investigate two-phase flow hydrodynamics during a steam generator tube rupture. The facility is 6.2 m high with a 0.5 x 0.5 m square cross-section in the test section. This has two staggered WMS with 3.4 mm spatial resolution to measure the bubble size and velocity. In addition, high speed cameras, PIV and LDV can be used to measure the velocity flow field.

5.12 Westinghouse in Sweden and US

Westinghouse Electric Company[22] has over 40 years' experience in thermal hydraulic testing of PWR and BWR fuel assemblies, and conducts thermal hydraulic tests at the following sites:

- Westinghouse Hot Cell Facility and Laboratories in Churchill, Pennsylvania (US);
- Westinghouse Columbia Fuel Fabrication Facility in Columbia, South Carolina (US); and
- Westinghouse Thermal-Hydraulic Test facility in Västerås, Sweden.

Westinghouse undertakes thermal hydraulics tests to investigate DNB, corrosion and crud, as well as LOCA and severe accident conditions [67]. Therefore, the current test facilities are focussed on heat transfer and flow-induced vibration related tests, and are accessible to use by other organisations.

5.12.1 Hot Cell Facility and Laboratories

The Hot Cell Facility and Laboratories in Churchill specialises in materials and chemistry evaluations through laboratory testing of both unirradiated and irradiated samples [68]. The site includes: five hot cells, extensive autoclave facilities, microstructural characterization laboratories, analytical chemistry laboratories, machining capabilities and mechanical testing laboratories. The thermal hydraulic test facilities include:

- WALT (Westinghouse Advanced Loop Tester) began operation in 2005 and was upgraded in 2015. It is used to study the impact on fuel thermal hydraulics due to crud deposited on the surface of PWR fuel rods. This includes crud deposition experiments (fuel clad temperatures) at normal operating conditions and different plant chemistry, as well as CHF tests with and without crud deposition and LOCA.
- The WATCH (Westinghouse Annular Thermal Crud Hydraulic) loop is used to understand and simulate crud behaviour at BWR operating conditions. This includes tests to validate the friction correlations and pressure drop on a crudded fuel rod, as well as measurements of fuel rod heat transfer enhancement due to 3D surface roughness and LOCA.

5.12.2 Columbia Fuel Fabrication Facility

There are three thermal hydraulics test facilities located in the Product Engineering Development Laboratory of the Columbia Fuel Fabrication Facility:

- The FACTS (Fuel Assembly Compatibility Test System) test facility was installed in 1988. It is an isothermal, deionised water loop to investigate pressure drop, FIV and debris mitigation on a single full-scale fuel assembly. The rod vibration is measured using custom built accelerometer rods, while the overall assembly vibration is measured using inductive displacement transducers in the test vessel walls [69].
- The VIPER (Vibration Investigation and Pressure drop Experimental Research) facility was installed in 1999. It is a deionised water loop to investigate pressure drop, FIV and grid-to-rod fretting performance on single or dual full-scale base and shorter height fuel assemblies [69].
- The VISTA (Vibration Investigation of Small-Scale Test Assemblies) loop was installed in 1999. This is used to test the hydraulic characteristics (pressure drop and FIV) of fuel grid designs using small-scale fuel bundles e.g. 5x5 at low temperatures and pressures. This includes a laser vibrometer to measure vibration [69].

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[22] http://www.westinghousenuclear.com/About/Innovation/Westinghouse-Lab-Network

5.12.3 Thermal-Hydraulic Test Facility in Västerås

Westinghouse Electric Sweden operates the following advanced thermal hydraulic test loops for BWR and PWR fuel testing [70]:

- FRIGG began operation in the 1960’s and the loop was modernised and upgraded in 1995. This BWR loop is used for both steady state and transient measurements of CHF (dryout) and pressure drop, as well as hydraulic stability and void fraction. This facility can test full-length, full-scale (8x8 to 11x11) and 1/4 fuel bundles using a variable axial and radial power profile with a maximum power of 15 MW. Production flow channels are used as the facility has “indirect” heater rods with thermocouples as dryout detectors. In addition, void fraction measurements are made using optical probes.
- BURE is a BWR loop for vibration and endurance testing (fuel rod fretting). This deionised water loop is used to investigate FIV and fretting on full bundles with pellets.
- FRODE is a BWR loop for single phase pressure drop measurements and debris catching tests at low pressure. This deionised water loop is used to test actual fuel bundles up to 2 m long to investigate flow distribution and lift forces, as well as debris catching and filter performance.
- ODEN began operation in 2010 after the Heat Transfer Research Facility (HTRF) in Columbia University closed in 2003. This is a PWR loop that is used for full-length, rod bundle CHF (DNB), pressure drop and mixing (thermal diffusion coefficient) tests. The ODEN facility is compatible with PWR fuel licensing requirements and uses directly heated rods (up to 6x6 arrays) with a maximum power of 12 MW. Measurements include high precision RTDs, flow meters, pressure taps and power sensors.

5.13 Texas A&M University in US

The Thermal-Hydraulic Research Laboratory is part of the Department of Nuclear Engineering within Texas A&M University (TAMU) in the US[23]. Research is currently undertaken on Light Water Reactors, Advanced Reactors and High Temperature Gas-cooled Reactors [71].

The test facilities occupy an area of 1,300 m^2, with 5 full-time professionals and over 30 researchers (PhD, MSc and undergraduates). It has 2 machine shops and an internal quality assurance program that conforms to 10 CFR 50 Appendix B and ASME NQA-1.

The TAMU Thermal-Hydraulic Research Laboratory is experienced at performing detailed velocity, temperature and void fraction measurements using the following instrumentation:

- Velocity - PIV (2D, stereoscopic and tomographic) and LDV systems.
- Temperature - DTS and LIF.
- Void fraction - Optical sensors and X-ray tomography.

The laboratory operates a number of test facilities [71], which include:

- The LWR CHF test facility has a dedicated AC and DC power supply (0.5 MW) and cooling water source with a maximum operating pressure of 3.5 MPa. This supports different operating conditions and customisable bundle configurations e.g. 5x5 rod bundle.
- Wire wrapped fuel assembly for advanced reactor technology, which is a large transparent test facility with 61 wire-wrapped clear acrylic pins and a fully accessible optically-clear test section. This enables high resolution velocity measurements with customisable axial and azimuthal pressure tappings for laminar, transition and turbulent regimes using different working fluids.
- The PTS (Pressurised Thermal Shock) test facility is used to investigate mixing of cold and hot water in a horizontal cold leg of a PWR. This includes high resolution measurements of velocity, concentration and pressure drop at various locations.
- The helical coil SG test loop includes three test facilities to investigate different aspects of the shell side flow. This includes PIV, LDA and hot film anemometry to measure the flow properties, as well as dynamic pressure transducers and PSP to measure pressure variation and pressure drop.
- Water-cooled RCCS (Reactor Cavity Cooling System) test facilities for High Temperature Gas-cooled Reactors. The water-cooled RCCS facility is 1:23 scale with nine risers to investigate natural convection in the RCCS. This includes customisable inlet conditions and high-resolution measurements, including PIV, LDV and DTS.

The TAMU Thermal-Hydraulic Research Laboratory has also generated high-fidelity experimental data for a number of international benchmarks, including:

- The ASME Verification and Validation in Computational Nuclear System Thermal Fluids Behaviour Committee (ASME V&V30 Standard Committee) Benchmark Problem 1 - Twin Jet CFD.
- CSNI 5th Benchmark of CFD applications for Nuclear Reactor Safety - The cold leg mixing CFD-UQ benchmark using the PTS test facility.

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[23] https://thrlab.tamu.edu/

6 Gas Facilities

The section provides an overview of the thermal hydraulic facilities relevant to Gas-Cooled Reactors (GCRs) i.e. HTGR, VHTR and GFR, which all use helium as a coolant. Helium has the benefit that it is chemically inert (allowing high temperature operation without corrosion) and single phase.

Significant research is being undertaken in China, EU, Japan, Korea, Russia, UK and US on a number of demonstrator projects. For example:

- The HTR-PM is a small modular nuclear reactor under development in China. It is a HTGR pebble-bed Gen IV reactor partly based on the earlier HTR-10 prototype reactor. The first demonstration plant is expected to start operating in 2019.
- The goal of the ALLEGRO project is to design, build and operate the first GFR demonstrator. The original concept of ALLEGRO was designed in France by CEA with the aim to develop a high temperature (850°C) fast flux test facility. The ALLEGRO project is being delivered by MTA EK (Hungary), ÚJV Řež, a.s. (Czech Republic), VUJE a.s. (Slovak Republic) and CEA (France), and the conceptual design is expected to be completed by 2025.

In 2010, a 5 year large-scale integrated research project THINS (Thermal-Hydraulics of Innovative Nuclear Systems) program was funded under FP7-EURATOM-FISSION European Commission. The project consisted of 23 partners from Europe and 1 partner from US [50]. A set of thermal hydraulic experimental facilities was designed and operated within the THINS project, of which 3 out of the 8 facilities related to VHTR designs:

- L-STAR, KIT;
- HOMER, PSI; and
- GP Loop, HZDR.

A Task group on Advanced Reactor Experimental Facilities (TAREF) was setup by CSNI to identify the research needed for GCRs and SFRs and recommend a path forward. CSNI initiated TAREF to provide an overview of facilities suitable for carrying out the safety research necessary for GCRs and SFRs. As a result, TAREF reviewed the experimental facilities for gas-cooled reactor safety studies, NEA/CSNI/R(2009)8 [72]. In addition, other experimental thermal hydraulic facilities related to GCRs have been identified from papers, conferences and publications.

Table 6 in Annex A provides a summary of GCR facilities ordered by country, and includes: organisation, maximum power, coolant, reactor type, status, test type and operating temperature and pressure.

The following research institutes and organisations that specialise in gas test facilities have been reviewed in more detail:

- INET in China (Section 6.1)
- ANL in US (Section 6.2)

The current research focus and areas of expertise are highlighted for each facility, together with their instrumentation and measurement capability and a description of the operational test rigs. The main parameters for each test rig are included in Table 6 in Annex A.

6.1 INET in China

The Institute of Nuclear and New Energy Technology (INET) at Tsinghua University in China was established in 1960[24]. INET began research into HTGRs in 1986, began construction of HTR-10 in 1995 and achieved criticality in 2000. HTR-10 is a 10 MW graphite-moderated helium gas-cooled reactor-test module. The design of the 200 MW HTGR Pebble-bed Module (HTR-PM) demonstration power plant began in 2004 and is currently going through construction with completion expected in 2019.

The INET division of Thermal Hydraulics undertakes research on the Nuclear Heating Reactor (NHR) and HTGR projects. This involves experimental studies on two-phase flow and flow instability for NHR, as well as helium studies for the HTR-PM.

The HTR-PM engineering laboratory was completed in 2011 to provide research and development test facilities to support the design, development and verification of the main components of the HTR-PM. This facility includes the following thermal hydraulic test facilities:

- The ETF-HT (Engineering Test Facility – Helium Technology) is a large scale helium loop to provide helium at reactor conditions (7 MPa, 750°C) [73]. The facility is 25 m long, 36 m wide and sunk 10 m into the ground. This heat source has been used to verify the performance of a full-scale steam generator assembly (ETF-SG), full-scale helium circulator (ETF-HC) and other systems.
- The TF-PBEC (Test Facility - Pebble-Bed Equivalent Conductivity) is used to measure the pebble-bed equivalent conductivity at 1600°C in a vacuum [74]. This is a full-scale heat transfer test facility that is 3 m diameter with 70,000 60 mm graphite spheres. The facility includes 90 high temperature thermocouples distributed throughout the annular domain.
- The TF-PBF3D (Test Facility - Pebble-Bed Flow 3D) is used to test the three-dimensional flow through the pebble-bed. This is a 1:5 scale facility, which uses air at atmospheric conditions.
- The reactor outlet mixing test facility investigates the performance of the thermal mixing structure at the HTR-PM reactor outlet [75]. The test facility geometry has a scaling factor of 1:2.5 and uses air as the working fluid at 70°C and atmospheric pressure. Measurements includes vortex flow meters, pressure sensors and thermocouples.

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[24] https://www.tsinghua.edu.cn/publish/ineten/5671/index.html

6.2 ANL in US

The Nuclear Engineering Division of the Argonne National Laboratory (ANL) in Illinois has been conducting nuclear research for over 70 years[25]. ANL undertake analytical and experimental thermal hydraulics research on current and advanced nuclear reactor designs.

The ANL facilities are capable of running large scale tests up to 23 m high with 3 MW of power to each building [76]. This includes two 1000 m^3 cells for containment experiments up to 0.4 MPa with on-site machine shop, welders and radiography.

The thermal hydraulic test facilities for HTGRs include:

- The NSTF (Natural convection Shutdown heat removal Test Facility) is a ½ axial scale model of the RCCS for the Modular High Temperature Gas cooled Reactor (MHTGR) [77]. The facility is 26 m high with 12 riser ducts, which represents a 19.03° sector. The test program was compliant with ASME NQA-1 and ran from 2014 to 2016 to investigate the air-cooled natural convection performance of the RCCS. The heat was applied over a 6.7 m height and could be applied in 3 modes (constant heat flux, constant temperature or axial power profile). This facility has been used to generate high-resolution temperature data for CFD validation. Measurements include power, heat flux, temperature, flow rate, pressure and external meteorological conditions. In addition, hot wire anemometers were used to measure velocity and turbulence within the riser ducts, and a distributed fibre optic temperature sensing system was used to generate temperature measurements every 10 mm at 1 Hz over each 7.5 m fibre (two riser ducts with 11 fibres each). ANL are currently looking at modifying NSTF to investigate water-based natural convection cooling.
- The MAX fluid dynamics facility is used to generate high resolution data of jet mixing [78]. This facility has two hexagonal air jets (22°C and 70°C) injected vertically into a 1 m x 1 m x 1.7 m long glass tank. Measurements include PIV, LDV, IR and high-speed video, as well as a 51 mm vertically spaced distributed fibre optic temperature sensing system on the tank mid-plane to provide 10 mm spatial temperature resolution.

In addition, ANL has several liquid metal test facilities[26], which include:

- The ALEX (Argonne Liquid metal EXperiment) facility was constructed in 1984 to conduct research on liquid metal systems. This includes 2 large and 1 small liquid alkali metal containments to undertake experiments on lithium and sodium systems. For example, this includes a sodium plugging test loop to investigate plugging in narrow channels due to impurities. Instrumentation includes thermocouples, pressure transducers, electromagnetic flow meters and pulsed thermocouple level probes.
- The METL (Mechanisms Engineering Test Loop) is used to test small and medium scale components and systems e.g. flow meters, level sensors, fuel handling systems, etc in two test vessels (18” and 28” diameter). The facility includes 3 m^3 of reactor grade liquid sodium across 4 vessels.
- The SNAKE (S-CO_2, NA Kinetics Experiment) facility investigates the chemical interaction when supercritical CO2 is injected into a sodium pool through a 64 μm diameter nozzle. Instrumentation includes flow meters, thermocouples, distributed fibre optic temperature sensors, mass spectrometers and gas analysers.

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[25] https://www.ne.anl.gov/facilities/
[26] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

7 Sodium Facilities

The Liquid Metal-cooled Fast Neutron Systems (LMFNS) catalogue is a living database[27], which is an electronic version of Section 4 of the IAEA Nuclear Energy Series publication [79]. In the LMFNS catalogue, 79 sodium facilities were reported to be in operation, construction, or design. These facilities are used, primarily, in support of the development of SFRs.

All of the SFR facilities are located in 9 countries (China, France, Germany, India, Japan, Korea, Latvia, Russia, and the US) of which 57 are operational and available for industrial research. The remaining 22 are in stand-by or under design, construction or upgrade.

Experimental activities are grouped within the IAEA catalogue according to their main research field, of which Thermal Hydraulics is one of 8 categories. For each facility, the IAEA catalogue includes a facility profile that provides a technical description, completed and planned experimental campaigns, training activities and references.

There are 43 sodium facilities in the LMFNS database that have thermal hydraulics listed as their main application. A summary of each of these facilities is presented in Table 7 in Annex A organised by country, and includes the organisation, maximum power, coolant, status, test type and operating temperature and pressure.

The following research institutes and organisations that specialise in sodium test facilities have been reviewed in more detail:

- CEA in France (Section 7.1)
- KIT in Germany (Section 7.2)

The current research focus and areas of expertise are highlighted for each facility, together with their instrumentation and measurement capability and a description of the operational test rigs. The main parameters for each test rig are included in Table 7 in Annex A.

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[27] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

7.1 CEA in France

CEA (Commissariat à l'énergie atomique et aux énergies alternatives) in Cadarache, France is leading the French research on Gen IV technologies[28], which is mainly focussed on SFRs and GFRs (to a lesser extent). In particular, CEA is the contracting authority for ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration), which is in the conceptual design phase.

ASTRID is a 600 MWe sodium-cooled fast reactor that was launched in 2010 with a core that is designed so that the reactivity will drop in the case of a sodium leak. CEA have four experimental platforms: CHEOPS, PAPIRUS, GISEH and PLINIUS [80] to support the ASTRID program. Descriptions of the CEA thermal hydraulic test facilities in these platforms are given in the LMFNS database[29] and described in Section 7.1.1 to 7.1.3, while thermal hydraulic facilities for other reactor technologies are briefly discussed in Section 7.1.4.

7.1.1 CHEOPS Experimental Platform

The CHEOPS (Circuits et Hall d'Essais des grOs comPosants en Sodium) experimental platform is currently being built with the first test expected in 2019. This is a large sodium facility (< 100 m^3) to test large scale components that has its own cleaning facility (STALACMITES). Instrumentation includes thermocouples, argon pressure measurement, inductive level probes, sodium flowmeters and pressure sensors. The coolant quality is actively measured and controlled with the impurities evaluated by a plugging indicator to less than a few ppm. The CHEOPS facility includes two sodium loops (NADYNE and NAIMMO) and the NSET loop.

- NADYNE is a test section (5 m high, 0.3 m diameter) with a small tank (10 tonnes) to allow the development of elements, such as the fuel assembly and safety components under steady-state and transient conditions.
- The NAIMMO test section (4 m high, 4.4 m diameter) is a large tank (70 tonnes) to develop and qualify large or full scale components, such as sealing joints, instrumentation and parts of the fuel handling system and control rod mechanisms under steady-state and transient conditions.
- The NSET facility will enable the development and verification of the sodium gas heat exchanger. The test section is 11 m high and includes a sodium loop (7 tonnes) and nitrogen loop. This will test full length heat exchanger modules at ½ power, and a full exchanger at a scale of 1:18 under steady-state and transient conditions.

7.1.2 PAPIRUS Experimental Platform

The PAPIRUS (PArc de Petites Installations de Recherche sur l'Utilisation du Sodium) experimental platform contains a number of small sodium loops (< 3 m^3) for thermal hydraulic code validation and component qualification. The test facilities within PAPIRUS are used to study corrosion, magnetohydrodynamics, instrumentation and small components immersed in sodium [80]. The coolant quality is actively measured and controlled to keep impurities less than a few ppm. The thermal hydraulic facilities in the PAPIRUS platform include:

- The DIADEMO Na facility is used to evaluate the performance of sodium/nitrogen heat exchanger mock-ups with 40kW power. In addition, it can test dynamic instrumentation, small components and experiments on Pb-Li / helium heat exchanger mock-ups in a 1.2 m

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[28] http://www.cea.fr/english/Pages/research-areas/nuclear-energy.aspx
[29] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

x 0.2 m x 0.2 m test section with 300 L of sodium and 55 L of Pb-Li. Instrumentation includes thermocouples, pressure sensors (sodium and gas), inductive level probes and electromagnetic sodium flow meters.

- The IRINA (Installation de Recherche pour l'Instrumentation en sodium NA) facility, which began operation in 1995 and was refurbished in 2011. It is a medium sized facility (1033 L sodium) for testing instrumentation, small components and thermal fatigue tests on mock-ups in a 2.7 m high and 0.75 diameter test section. Instrumentation includes thermocouples, argon pressure sensors, inductive level probes and electromagnetic flow meters.

7.1.3 GISEH Experimental Platform

The GISEH (Groupement des Installations en Simulant Eau pour l'Hydraulique) experimental platform uses water to undertake tests for component qualification and code validation of gas entrainment and hot pool flows. The thermal hydraulic facilities in the GISEH platform include:

- The PLATEAU (PLATeforme en EAU) test facility is a modular platform for thermal hydraulic tests on large scale mock-ups. For example, MICAS is a 360°, 1:6 model of the upper plenum that is tested under steady-state and transient conditions [81]. Instrumentation includes flow meters, thermocouples, optical fibre temperature sensors, high speed video, PIV and LDV.
- The BACCARA (Banc de Caractérisation d'Assemblages de Réacteurs Avancés) facility is used to study fuel assembly vibration and cavitation at 1:1 scale (3 m long test section). Instrumentation includes flow, temperature and pressure with local velocity measured using laser techniques through windows.

7.1.4 Other CEA Facilities

CEA also has a number of other thermal hydraulic facilities to undertake research on other reactor technologies, which include:

- Thermal hydraulic experiments to support GFR research for the ALLEGRO project, such as ESTHAIR, HECO, HEDYT, HELITE and SALSA [72].
- Thermal hydraulic facilities to support LWR research, which include containment studies (MISTRA and SULTAN) [39] and two-phase flow and vibration experiments (CLOTAIRE) [82].

7.2 KIT in Germany

The Karlsruhe Institute of Technology (KIT) in Germany was formed from the Universität Karlsruhe and Forschungszentrum Karlsruhe in 2009, and has over 50 years of nuclear research experience. KIT currently undertake research within several different institutes across a range of advanced reactor technologies.

7.2.1 Institute for Neutron Physics and Reactor Technology

The Institute for Neutron Physics and Reactor Technology (INR) in KIT[30] undertake experiments and theoretical research related to nuclear safety research and fusion, as well as the transformation and storage of heat using solar power. The INR experimental facilities include a number of sodium test facilities, which include:

- The KASOLA (KArlsruhe SOdium LAboratory) is used to investigate flow phenomena in liquid sodium, which includes turbulent and convective heat transfer and free surface flows. This is a flexible facility with a nearly 6 m high test section and 7 m^3 of liquid sodium to undertake thermal hydraulic benchmark experiments to validate CFD models. Instrumentation includes flow meters, thermocouples and UDV sensors.
- The SOLTEC (SOdium Loop to TEst materials and Corrosion) facility consists of three separate test loops (SOLTEC-1, SOLTEC-2 and SOLTEC-3) [83]. These are small scale facilities (14 L sodium) for material qualification and steel corrosion investigations, such as long term tests and low cycle fatigue.

Other INR test facilities include:

- L-STAR (Luft - STab, Abstandshalter, und Rauigkeiten), which means air - rod, spacer grid and roughness, incorporates two loops: a small loop (SL) and large loop (LL)[31]. This is used to investigate turbulent convective heat transfer in air for smooth and rough heated rod surfaces, which is relevant for HTGRs [50]. The test section is 3.7 m long and consists of a hexagonal tube with a heated rod inside. Instrumentation includes RTDs, thermocouples, flow meters and high precision pressure sensors. In addition, two windows in the test section allow high resolution measurements using LDA, PIV and LIF.

7.2.2 Institute for Nuclear and Energy Technologies

The Institute for Nuclear and Energy Technologies (IKET) in KIT[32] undertake experiments and theoretical research related to thermal hydraulic phenomena, including chemical reactions, magnetic fields and transport of gases, liquids, and solids. This includes liquid metal flows, combustion and explosions.

The KArlsruhe Liquid metal LAboratory (KALLA) operate several liquid metal test loops to understand the physics of flow and heat transfer in molten metals. In addition, KALLA is developing measurement equipment for liquid metal flows. The KALLA test facilities include:

- The ALINA (kArlsruhe experiments with LI and NA free jet) test facility began operation in 2007 and is currently on standby. This facility contains 150 L of sodium and was used to study liquid metal jets and free surface flows. Instrumentation includes flow meters and thermocouples.
- THEADES (THErmal hydraulics and Ads DESign) is a forced convection loop with 4 m^3 of LBE to investigate rod bundle heat transfer, free surface flows and heat exchanger

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[30] http://www.inr.kit.edu/english/index.php
[31] http://www.inr.kit.edu/english/256.php
[32] http://www.iket.kit.edu/english/index.php

performance in order to provide data for model validation [84]. The oxygen content is controlled using electrochemical oxygen sensors. Instrumentation includes flow meters, thermocouples and pressure transducers.

- The THESYS (Technologies for HEavy metal SYStems) test facility is used to investigate turbulent heat transfer in an annular gap, oxygen control and flow meter qualification. The facility contains 220 L of LBE and has a moveable heated rod to allow temperature measurements along the 2 m test section. Instrumentation includes flow meters, thermocouples and pitot tubes.

Other IKET test facilities include:

- COSMOS-L (Critical-heat-flux On Smooth and MOdified Surfaces - Low pressure loop) is a water-steam test loop to investigate convection boiling near CHF. This uses high speed cameras to understand the influence of surface structure on the microscopic boiling process.
- COSMOS-H (Critical-heat-flux On Smooth and MOdified Surfaces - High pressure loop) is being built to visualise boiling under prototypical reactor conditions for validation of CFD models. This will include a modular test section with thermocouples and pressure sensors and windows with optical access for PIV, LDA and high-speed imaging. The initial test programme will investigate a single heated rod in an annular gap.
- The WENKA (Water ENtrainment Channel KArlsruhe) facility investigates counter-current stratified two-phase flows on horizontal and bent channels. This is intended to provide high-resolution velocity and void fraction data for CFD model validation. Instrumentation includes PIV, optical void measurement, high-speed video and phase doppler anemometry.
- HYKA (Hydrogen Test Centre) is run by the Flow and Combustion Engineering team[33] within IKET [39] [85]. This test centre investigates hydrogen distribution, combustion and detonation using a range of test vessels. This includes the A1-Vessel (98 m^3), A3-Vessel (30 m^3) and A6-Vessel (21.5 m^3) for combustion experiments, a 160 m^3 flow test chamber for hydrogen release experiments and two explosion tubes. Instrumentation includes thermocouples, piezoelectric pressure gauges, sonic hydrogen sensors, photodiodes for velocity and strain gauges. In addition, the flow test chamber has high speed cameras and laser displacement sensors.

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[33] https://www.iket.kit.edu/english/345.php

8 Lead Facilities

The Liquid Metal-cooled Fast Neutron Systems (LMFNS) catalogue is a living database[34], which is an electronic version of Section 4 of the IAEA Nuclear Energy Series publication [79]. In the LMFNS catalogue, 72 lead facilities were reported to be in operation, construction, or design. These facilities are used, primarily, in support of the development of LFRs, which includes all heavy liquid metal cooled reactors, i.e. lead and Lead Bismuth Eutectic (LBE) cooled.

All of the LFR facilities are located in 13 countries (Belgium, China, Czech Republic, France, Germany, Latvia, Italy, Spain, Japan, Korea, Sweden, Russia, and the US) of which 56 are operational and available for industrial research. The remaining 16 are in stand-by or under design, construction or upgrade.

Experimental activities are grouped within the IAEA catalogue according to their main research field, of which Thermal Hydraulics is one of 8 categories. For each facility, the IAEA catalogue includes a facility profile that provides a technical description, completed and planned experimental campaigns, training activities and references.

There are 33 lead facilities in the LMFNS database that have thermal hydraulics listed as their main application. A summary of each of these facilities is presented in Table 8 in Annex A organised by country, and includes the organisation, maximum power, coolant, status, test type and operating temperature and pressure.

The following research institutes and organisations that specialise in lead test facilities have been reviewed in more detail:

- SCK•CEN in Belgium (Section 8.1)
- ENEA in Italy (Section 8.2)

The current research focus and areas of expertise are highlighted for each facility, together with their instrumentation and measurement capability and a description of the operational test rigs. The main parameters for each test rig are included in Table 8 in Annex A.

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[34] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

8.1 SCK•CEN in Belgium

SCK•CEN (StudieCentrum voor Kernenergie • Centre d'etude de l'Energie Nucléaire - Belgian Nuclear Research Centre) was established in Mol in 1952[35]. The Institute for Advanced Nuclear Systems, one of 3 institutes, is focussed on the development of innovative nuclear reactors. This includes the design and build of a new research reactor, MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) [86].

The MYRRHA research project began in 1999, and currently involves 20 research centres across Europe and 30 universities. It is the first prototype reactor driven by a particle accelerator, known as an Accelerator Driven System (ADS), which means that the reactor core is 'subcritical' and so inherently safe. The construction of MYRRHA was approved by the Belgian government in 2018, and will enable the further development of lead fast reactors. SCK•CEN have designed and built a number of thermal hydraulic test facilities as part of the MYRRHA design and development process, which include:

- COMPLOT (COMPonent LOop Tests) is a large scale loop that is used for full scale performance and hydraulic tests of MYRRHA core components with 9 tonnes of LBE. The facility includes two 4" diameter test sections (3 m and 12 m long). Tests include pressure drop and FIV for a fuel assembly, control and safety rods operational tests and spallation target pressure drop and velocity profile at the beam window. Instrumentation includes vortex flow meters, UDV sensors, remote seal diaphragm pressure transmitters, level sensors and fibre Bragg grating (fibre optic sensors).
- E-SCAPE (European SCAled Pool Experiment) is a 1/6th scale model (2 m high, 1.4 m diameter) of the MYRRHA reactor. It is designed to simulate the main thermal hydraulic phenomena using an electrically heated core with 27 tonnes of LBE. This will investigate the natural convection decay heat removal via heat exchangers and residence times in the whole system, upper plenum thermal mixing, stratification, flow distribution and free surface oscillation, lower plenum flow distribution and pump jet behaviour. The facility includes a large number of in-vessel measurements, which include UDV velocity probes, thermocouples, pressure sensors, radar level sensors and Coriolis flow rate sensors.

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[35] http://sckcen.be/en

8.2 ENEA in Italy

The ENEA (National Agency for New Technologies, Energy and Sustainable Economic Development) has an extensive nuclear fission programme to undertake a range of research and development opportunities including; advanced nuclear systems, research reactors, characterisation of reactor material, characterisation of nuclear materials and radioactive waste, radioactive waste management, and modelling and simulations[36].

The ENEA Brasimone Research Centre[37] near Bologna was set up in the 1960s to carry out studies on fast breeder European reactors. The centre employs around 130 people and conducts thermal hydraulic research on fusion technologies and lead-cooled fast reactors. The thermal hydraulic test facilities for LFR development include:

- The CIRCE (CIRCulation Eutectic) facility consists of a 8.5 m high (1.2 m diameter) cylindrical vessel with 70 tonnes of liquid LBE. The current ICE (Integral Circulation Experiment) configuration is used to test the performace of the HERO (Heavy liquid mEtal pRessurized water cOoled tube) heat exchanger. The facility is used to investigate thermal stratification and mixing in a lead pool under forced and natural circulation conditions. In addition, tests have been conducted on a fuel rod bundle (37 pins) to study heat transfer and cladding temperatures. The facility includes an argon cover gas, oxygen monitoring and control system and two filtering devices. Instrumentation includes thermocouples, pressure transducer, oxygen sensors, flow meters, laser level gauge and hot wire anemometers.
- The CIRCE-SGTR (Steam Generator Tube Rupture) experiments involved modifying the CIRCE cover vessel to support the dedicated SGTR test section. This was designed to undertake SGTR experiments relevant to the MYRRHA reactor to investigate the impact of water injection into the LBE pool during a water tube rupture. Different locations for the water tube rupture will be studied with instrumentation that includes thermocouples, pressure transducers, venturi flow meters, bubble tubes and strain gauges.
- HELENA (HEavy Liquid metal Experimental loop for advanced Nuclear Applications) is a multipurpose facility for LFR development using liquid lead. The facility has been designed with a number of different test sections to qualify centrifugal pumps, structural materials for erosion/corrosion, valves, heat exchangers and instruments. In addition, there is a heating section to investigate heat transfer on wire wrapped fuel bundles. The oxygen content is measured and controlled and instrumentation includes thermocouples, pressure transducers and venturi boccaglio flow meters.
- The LIFUS5 (LIthium FUSion) facility is designed to operate with different heavy liquid metals (Lead, Lithium-Lead alloy and LBE). The test section is 0.85 m high and 0.54 m diameter and used to investigate the interaction with water during a SGTR event and determine the leak rate and bubble size through typical cracks. Instrumentation includes thermocouples, pressure transducers, flow meters and strain gauges.
- NACIE-UP (NAtural CIrculation Experiment UPgrade) is a rectangular loop to research natural circulation using lead or LBE. The loop is 8 m high with a 2.2 m high test section and simulated fuel bundle (19 pins) in the bottom of the riser which is heated over 0.6 m. This is used to study heat transfer and natural circulation behaviour of the MYRRHA fuel assembly during a Loss of Flow Accident. Instrumentation includes thermocouples, pressure transducers and induction flow meters.

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[36] http://www.enea.it/en/research-development/nuclear-energy/nuclear-fission
[37] http://old.enea.it/com/ingl/center/Brasimone/index.html

- SIRIO (SIstema di RImozione della potenza di decadimento per reattori nucleari innOvativi) meaning decay heat removal system for innovative nuclear reactor. This facility uses pressurised water to investigate the behaviour of the decay heat removal system and ALFRED isolation condenser. This uses a steam generator and in-pool condenser with non-condensable gas injection to investigate the impact of non-condensable gas on heat transfer rate. Instrumentation includes thermocouples, pressure transducers and mass flow meters.
- The SOLIDX (SOLIDification eXperiment facility) facility is used to study freezing and de-freezing of lead in a pool. The test section has a length of 0.35 m and diameter of 0.273 m. Two heaters are located asymmetrically in the pool with a water jacket around the outside to undertake freezing and de-freezing experiments with the temperature measured using thermocouples at various locations.

9 Molten Salt Facilities

In May 2004, a Provisional System Steering Committee (PSSC) within the GIF Policy Group was set up to further the cooperation on MSRs. The collaborating committee includes EURATOM, France and the Russian Federation. USA, China, Japan and South Korea are observers within the MSR-PSSC.

Even though the MSR concepts are focused on different baseline concepts; Molten Salt Fast Reactor (MSFR), MOlten Salt Actinide Recycler and Transmuter (MOSART), Thorium Molten Salt Reactor (TMSR) and Fluoride-salt-cooled High-temperature Reactors (FHR), large commonalities in basic research and development areas exist and the Gen IV framework is vital to optimise the R&D effort [87]. Some of the molten salt research programs that are underway around the world include:

- **China:** Since 2011, the Chinese Academy of Sciences (CAS) have been working on the TMSR Nuclear Energy System (NES). One of the main aims of this project is to develop solid and liquid-fuelled MSRs which strive for realising effective Thorium energy utilisation and hydrogen production by nuclear energy within 20-30 years [87].
- **European Union:** New research programmes have been launched on further performances and design parameters of MSR concepts. These projects are within the EURATOM 5th, 6th and 7th Framework Programs (MOST, ALISIA and EVOL projects). These activities are in close collaboration with complimentary International Science & Technology Centre projects (ISTC #1606 and #3749) in the Russian Federation on MSRs. In addition, SAMOFAR[38] (Safety Assessment of the Molten Salt Fast Reactor) is one of the major Research and Innovation projects in the Horizon 2020 EURATOM research programme.
- **Japan:** With over 30 years of work experience on MSRs, Furukawa and his group proposed a THORIum Molten Salt Nuclear Energy Synergetic (THORIMS-NES) system in the 1980s [88], [89]. THORIMS-NES is a combination of fission power reactor of Molten Salt Reactor (MSR-FUJI), and Accelerator Molten Salt Breeder (AMSB) for production of fissile 233U [87]. Besides the design study, several molten salt loops were manufactured by this group.
- **USA:** The US programs on MSRs focuses on both MSRs operating with liquid fuel, and those operating with solid-fuel (also known as FHRs) [87]. The Department of Energy's (DOE) focused investment in FHRs is through University Research with university led integrated research projects ($5M each) focused on addressing technical issues for FHRs initiated from 2015 to 2018. In addition, the USA (ORNL) and China (SINAP) have begun cooperating on FHR research through a Cooperative Research and Development Agreement (CRADA).

A summary of the facilities that have been identified is presented in Table 9 in Annex A organised by country, and includes the organisation, maximum power, coolant, status, test type and operating temperature and pressure.

The following research institutes and organisations that specialise in molten salt test facilities have been reviewed in more detail:

- LPSC in France (Section 9.1)
- INL in US (Section 9.2)

The current research focus and areas of expertise are highlighted for each facility, together with their instrumentation and measurement capability and a description of the operational test rigs. The main parameters for each test rig are included in Table 9 in Annex A.

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[38] http://samofar.eu/

9.1 LPSC in France

LPSC (Laboratoire de Physique Subatomique et de Cosmologie) in Grenoble is affiliated to IN2P3 (Institut National de Physique Nucléaire et de Physique des Particules) and CNRS (Centre National de la Recherche Scientifique).

CNRS has been researching molten salt reactors since 1997[39]. The MSFR (Molten Salt Fast Reactor) team within the Physics for Energy and Health group at LPSC has been undertaking thermal hydraulics studies on molten salt as part of the SAMOFAR Horizon 2020 project. The LPSC thermal hydraulic facilities include:

- The FFFER (Forced Fluoride Flow for Experimental Research) facility is a forced convection loop using FLiNaK molten salt with argon gas injected into the loop [90]. This is used to study the performance of a bubble separator and investigate the bubble separation process within molten salt. The loop also includes a cold plug device, which is designed to melt during a loss of power fault. The performance of the cold plug device can be investigated within the facility to study how quickly the plug melts in the case of a loss of power.
- The SWATH (Salt at WAll: Thermal ExcHanges) experiment is part of the European Horizon 2020 SAMOFAR project [90]. This is intended to improve molten salt numerical methods and develop the design and principle of the cold plug device. This is achieved by using a controlled flow through simple test section geometries. Two facilities will be built SWATH-W, which uses water as the working fluid and SWATH-S, which uses FLiNaK molten salt. SWATH-W is manufactured with Plexiglas and uses water at ambient conditions, which allows detailed PIV measurements of the flow through the test sections. The SWATH-S test sections will be designed to investigate molten salt phase change and heat exchange experiments in open and closed channels. This will provide data for solidification, turbulence and radiative heat transfer models.

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[39] http://lpsc.in2p3.fr/index.php/en/groupes-de-physique/enjeux-societaux/msfr

9.2 INL in US

INL (Idaho National Laboratory) is one of the US Department of Energy's National Laboratories, and is a centre for nuclear energy research and development[40]. INL undertake research into new fuels and materials, reactor systems and waste management options, as well as managing strategic partnerships such as the GAIN (Gateway for Accelerated Innovation in Nuclear)[41] initiative.

INL have conducted thermal hydraulic experiments related to nuclear power for over 70 years, and continue to build and operate a number of thermal hydraulic test rigs, which include:

- The ARTIST (Advanced Reactor Technology Integral System Test) facility is currently being constructed at the Energy Systems Laboratory at INL [91]. This is a high-temperature multi-loop (water, helium and molten salt) test facility to support thermal hydraulic, materials and thermal energy storage research. The water flow loop (15 MPa, 325 °C) will be built first to evaluate new cladding materials, advanced heat exchanger designs, FIV of fuel rod bundles and natural circulation studies under prototypical conditions. The helium and molten salt loops will be used to investigate flow and heat transfer issues. To support the ARTIST facility, INL have conducted experiments to gain experience of fluoride salt mixture handling, preparation and purification and measure salt mixture thermophysical properties.
- The MIR (Matched-Index-of-Refraction) flow facility[42] has a large test section (0.6 m square, 2.4 m long) that enables high spatial and temporal resolution measurements for CFD validation [92]. The working fluid is temperature-controlled mineral oil with models made from fused quartz to eliminate optical refraction i.e. the model is invisible. This allows non-intrusive velocity measurements using a 3-D PIV system, LDV and particle tracking velocimetry. Experimental studies have included a prismatic reactor core model, nuclear reactor lower plenum, fuel rod channels and spent nuclear fuel storage canisters, as well as airflow around buildings, boundary layer transition and boundary layer over turbine blade models.

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[40] https://inl.gov/research-programs/nuclear-energy/
[41] https://gain.inl.gov/SitePages/Home.aspx
[42] https://mir.inl.gov/

10 Conclusions

A definitive review of the state-of-the-art of worldwide thermal hydraulic research facilities with relevance to Small Modular Reactor and Generation IV nuclear reactor designs has been undertaken. This includes thermal hydraulic test facilities associated with molten salt, lead, sodium, gas and water cooled reactor designs

This report provides a summary of the thermal hydraulic test facilities organised by country, and highlights some of the key parameters associated with each test facility. This demonstrates that significant experimental testing has been conducted across all Generation III, Small Modular Reactor and Generation IV reactor technologies throughout the world, particularly in Canada, China, France, Germany, India, Italy, Japan, Korea, Switzerland, Russia and USA.

Although this report does not provide a definitive, exhaustive list, the number of test facilities for each reactor technology gives an indication of the extent of the international research:

- 87 water (PWR, BWR and SCWR) reactor SET facilities for natural circulation, two-phase flow, flow-induced vibration and component flow phenomena;
- 34 water (PWR and BWR) reactor containment facilities for combustion, condensation, aerosol and non-condensable gas behaviour;
- 24 gas (HTGR, VHTR and GFR) thermal hydraulic test facilities;
- 42 sodium (SFR) thermal hydraulic test facilities;
- 31 lead (LFR) thermal hydraulic test facilities; and
- 16 molten salt (MSR) thermal hydraulic test facilities.

This review demonstrates that there are a large number of successful nuclear thermal hydraulic test facilities around the world with comprehensive coverage of phenomena and reactor types. Therefore, any new facility will need to be carefully planned in order to make a major impact

Test facilities in a few organisations have been reviewed in more detail with a focus on ‘facility’ capability rather than test rig capability, in order to better understand the current state-of-the-art in thermal hydraulic test facilities. These examples highlight how existing test facilities have developed a strong, distinctive capability within nuclear thermal hydraulics with their own individual areas of research, measurement techniques or specific expertise, with research programmes spanning decades.

In the past, experimental facilities have generally focused on tests that resulted in data more suitable for safety assessments and the validation of system codes. However, current reactor developments require more extensive validation data to develop more efficient and less over-engineered designs in shorter development periods. This requires more detailed data from extensive experimental datasets and the use of advanced diagnostic techniques.

The number and breadth of experimental facilities that are currently built and operated around the world shows that the international nuclear community recognises the need for experimental research and testing to support nuclear reactor development. There is consequently room for the UK to play a part in meeting these needs. However, in order for the UK facility to be successful, it needs to develop its own unique expertise and distinctiveness. One strong possibility is in high quality novel measurement and diagnostic techniques to generate data of sufficient quality to be used for CFD model validation.

Abbreviations

AHTR	Advanced High Temperature Reactor
AMSB	Accelerator Molten Salt Breeder
ARIS	Advanced Reactor Information System
BDBA	Beyond Design Basis Accident
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
CFD	Computational Fluid Dynamic
CHF	Critical Heat Flux
CRADA	Cooperative Research and Development Agreement
DBA	Design Basis Accident
DP	Pressure Drop
DEC	Design Extended Condition
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling Systems
EPR	European Pressurised Reactor
ESBWR	Economic Simplified Boiling Water Reactor
FHR	Fluoride-salt-cooled High temperature Reactor
FIV	Flow Induced Vibration
FSI	Fluid-Structure Interaction
GCR	Gas Cooled Reactor
Gen-IV	Generation IV
GFR	Gas-cooled Fast Reactor
HTC	Heat Transfer Coefficient
HTGR	High-Temperature Gas-cooled Reactor
HWR	Heavy Water Reactor
ICS	Isolation Condenser System
IET	Integral Effect Test
LBE	Lead Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
LMFNS	Liquid Metal-cooled Fast Neutron Systems
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MASLWR	Multi-Application Small Light Water Reactor
MOSART	MOlten Salt Actinide Recycler and Transmuter
MSR	Molten Salt Reactor
MSFR	Molten Salt Fast Reactor
NPP	Nuclear Power Plant
PCCS	Passive Containment Cooling System
PHWR	Pressurised Heavy Water Reactor

PSS	Pressure Suppression System
PTS	Pressurised Thermal Shock
R&D	Research and Development
RCCS	Reactor Cavity Cooling System
SA	Severe Accident
SAMOFAR	Safety Assessment of the MOlten salt FAst Reactor
SCWR	SuperCritical Water-cooled Reactor
SET	Separate Effect Test
SFR	Sodium-cooled Fast Reactor
SMR	Small Modular Reactor
TAREF	Task Group on Advanced Reactor Experimental Facilities
TDC	Thermal Diffusion Coefficient
TH	Thermal Hydraulics
THINS	Thermal Hydraulics of Innovative Nuclear Systems
TIETHYS	The International Experimental Thermal HYdraulics Systems
THORIMS-NES	THORIum Molten Salt Nuclear Energy Synergetic
TMSR	Thorium Molten Salt Reactor
UDV	Ultrasound Doppler Velocimetry
VHTR	Very High Temperature Reactor
VVER	Water-Water Energetic Reactor

11 Organisations

AAEC	Australian Atomic Energy Commission (Australia)
AECL	Atomic Energy of Canada Limited (Canada)
ANL	Argonne National Laboratories (USA)
ARC	Advanced Reactor Concepts (USA)
BARC	Bhabha Atomic Research Centre (India)
BATTELLE	Battelle Memorial Institute (USA)
BEIS	Department for Business, Energy and Industrial Strategy (UK)
BT	Becker Technologies (Germany)
B&W	Babcock & Wilcox (USA)
CAER	Centre for Advanced Engineering Research (USA)
Carleton	Carleton University (Canada)
CAS	Chinese Academy of Sciences (China)
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives (France)
CGNPC	China General Nuclear Power Corporation (China)
CIRTEN	Consortium of Italian universities (Italy)
CIAE	China Institute of Atomic Energy (China)
CISE	Centre for Information, Study and Experience (Italy)
CNEIC	China Nuclear Energy Industry Corporation (China)
CNL	Canadian Nuclear Laboratories (Canada)
CNNC	China National Nuclear Corporation (China)
CNRS	Le Centre National de la Recherche Scientifique (France)
CSNI	NEA Committee on the Safety of Nuclear Installations
CVNP	Carolinas Virginia Nuclear Power Association (USA)
DCNS	French shipbuilder (now Naval Group, France)
DOE	Department of Energy (USA)
ENEA	National Agency for New Technologies, Energy & Sustainable Economic Development (Italy)
EREC	Electrogorsk Research and Engineering Center on NPP safety (Russia)
EURATOM	European Atomic Energy Community (EU)
FURGS	Federal University of Rio Grande do Sul (Brazil)
FZJ	Forschungszentrum Jülich (Germany)
GE	General Electric (nuclear company, USA)
GIDOPRESS	OKB Gidopress nuclear company (Russia)
GIF	Generation IV International Forum
GKSS	Society for the Utilisation of Atomic Energy in Shipbuilding and Shipping Ltd (Germany)
HEDL	Hanford Engineering Development Laboratory (USA)
HZDR	Helmholtz-Zentrum Dresden-Rossendorf (Germany)
IAEA	International Atomic Energy Agency
IGCAR	Indira Gandhi Centre for Atomic Research (India)
INEST	Institute of Nuclear Energy Safety Technology (China)

INET	Institute of Nuclear and New Energy Technology, Tsinghua University (China)
INL	Idaho National Laboratory (USA)
IPPE	I.I. Leypunsky Institute of Physics and Power Engineering (Russia)
IPUL	Institute of Physics of University of Latvia (Latvia)
IRSN	Institute for Radiological Protection and Nuclear Safety (France)
ITMSF	International Thorium Molten Salt Forum
JAEA	Japan Atomic Energy Agency (Japan)
JAERI	Japan Atomic Energy Research Institute (Japan)
JRC	Ispra Joint Research Centre of the European Commission (Italy)
KAERI	Korean Atomic Energy Institute (Korea)
KFK	Kommission zur Überprüfung des Kernenergieausstiegs (Germany)
KIT	Karlsruhe Institute of Technology (Germany)
KTH	Royal Institute of Technology (Stockholm, Sweden)
KWU	Kraftwerk Union (nuclear company, Germany)
LPSC	Laboratory of Subatomic Physics & Cosmology (Grenoble, France)
Manitoba	University of Manitoba (Canada)
MARVIKEN	Nuclear reactor (Sweden)
MIT	Massachusetts Institute of Technology (USA)
Montreal	Polytechnique Montréal (Canada)
NIFS	National Institute for Fusion Science (Japan)
NIRAB	Nuclear Innovation and Research Advisory Board (UK)
NNI	Northern Nuclear Industries Incorporated (Canada)
NU	Northwestern University (USA)
NUTRECK	NUclear TRansmutation Energy research Center of Korea (Korea)
OECD	Organisation for Economic Co-operation and Development
OKBM	JSC Afrikantov OKB Mechanical Engineering (Russia)
ORNL	Oak Ridge National Laboratory (USA)
OSU	Oregon State University (USA)
PoliMi	Politecnico di Milano (Italy)
PSI	Paul Scherr Institute (Switzerland)
Purdue	Purdue University (USA)
RDIPE	Research and Development Institute of Power Engineering (Russia)
ROSATOM	State Atomic Energy Corporation (Russia)
RRC KI	Russian Research Centre “Kurchatov Institute” (Russia)
SCK-CEN	Belgian Nuclear Research Centre (Belgium)
SIET	Nuclear test laboratory (Italy)
SINAP	Shanghai Institute of Applied Physics (China)
SNERDI	Shanghai Nuclear Engineering Research and Design Institute (China)
SNU	Seoul National University (Korea)
SRI	Stanford Research Institute (USA)

Stern	Stern Laboratories (Canada)
STL	Steenkampskraal Thorium Limited (South Africa)
STUDSVIK	Swedish nuclear company (Sweden)
SVUSS	Statni Vyzkumny Ustav pro Stavbu Stroju (Czech Republic)
TAMU	Texas A&M University (USA)
Tokyo	University of Tokyo (Japan)
TOSHIBA	Toshiba Corporation (Japan)
TU Delft	Delft University of Technology (Netherlands)
TUM	Technical University of Munich (Germany)
UCB	University of California at Berkeley (USA)
uOttawa	University of Ottawa (Canada)
USC	University of South Carolina (USC)
UCSB	University of California Santa Barbara (USA)
UKAEA	UK Atomic Energy Authority (UK)
UM	University of Michigan (USA)
UNIPI	University of Pisa (Italy)
USNRC	United States Nuclear Regulatory Commission (USA)
UW-M	University of Wisconsin-Madison (USA)
VKI	Von Karman Institute for Fluid Dynamics (Belgium)
VTT	Technical Research Centre of Finland (Finland)
WEC	Westinghouse Energy Company (USA)
WGAMA	NEA/CSNI Working Group on Analysis and Management of Accidents
XJTU	Xi'an Jiaotong University (China)

12 References

[1] NIRAB-75-10, “UK Nuclear Innovation and Research Programme Recommendations,” March 2016.

[2] NIRAB-124-1, “Prioritisation of UK Nuclear Innovation and Research Programme Recommendations,” November 2016.

[3] “Industrial Strategy Nuclear Sector Deal,” Department for Business, Energy and Industrial Strategy, 2018. [Online]. Available: https://www.gov.uk/government/publications/nuclear-sector-deal.

[4] D. Bramble, “Nuclear Research and Development Programme: Invitation to Tender for R&D on Digital Nuclear Reactor Design,” BEIS, TRN 1210/09/2016(3), November 2016.

[5] “Technology Roadmap Update for Generation IV Nuclear Energy Systems, Generation IV International Forum,” https://www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf, January 2014.

[6] “Advances in Small Modular Reactor Technology Developments,” A Supplement to: IAEA Advanced Reactors Information System (ARIS), http://aris.iaea.org, September 2018.

[7] “A state-of-the-art report on scaling in system thermal-hydraulics applications to nuclear reactor safety and design,” NEA/CSNI/R(2016)4 https://www.oecd-nea.org/nsd/docs/2016/csni-r2016-4.pdf, March 2017.

[8] “Quality Management Systems - Requirements,” ISO 9001:2015, 2015, 5th Edition.

[9] “Nuclear Safety and Quality Management System - Requirements,” Nuclear Quality Standard Association, NQSA, NSG-100, 2014.

[10] “Quality management systems -- Specific requirements for the application of ISO 9001:2015 by organizations in the supply chain of the nuclear energy sector supplying products and services important to nuclear safety (ITNS),” ISO 19443:2018, 2018, 1st Edition.

[11] “Occupational health and safety management systems - Requirements,” ISO 45001:2018, 2018, 1st Edition.

[12] “General requirements for the competence of testing and calibration laboratories,” ISO 17025:2005, 2005, 2nd Edition.

[13] “Conformity assessment - Requirements for the operation of various types of bodies performing inspection,” ISO 17020:2012, 2012, 2nd Edition.

[14] “Quality Assurance Requirements for Nuclear Facility Applications,” American Society of Mechanical Engineers, ASME NQA-1-2008, New York, 2008.

[15] “Appendix B to Part 50 - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” NRC Regulations, Title 10, Code of Federal Regulations, 10 CFR 50 Appendix B, 2018.

[16] P. J. LaNasa and E. Upp, Fluid Flow Measurement (Third edition), Oxford: Butterworth-Heinemann, 2014.

[17] A. Morris and R. Langari, Measurement and Instrumentation (Second Edition), Boston: Academic Press, 2016.

[18] M. Jahanmiri, “Pressure Sensitive Paints: The Basics & Applications,” Chalmers University Library, Research Report 2011:07, June 2011.

[19] C. Gerardi, N. Bremer, D. Lisowski and S. Lomperski, “Distributed temperature sensor testing in liquid sodium,” *Nuclear Engineering and Design,* pp. Volume 312, Pages 59-65, 2017.

[20] K. Kim and S. Feng, Application of Thermo-Fluidic Measurement Techniques, Butterworth-Heinemann, 2016.

[21] T. Astarita and G. Carlomagno, Experimental Fluid Mechanics - Infrared Thermography for Thermo-Fluid-Dynamics, Berlin: Springer, 2013.

[22] T. Liu, “Pressure- and Temperature-Sensitive Paints,” in *Encyclopedia of Aerospace Engineering*, John Wiley & Sons Ltd, 2011.

[23] H. V. Peña and O. Rodriguez, “Applications of wire-mesh sensors in multiphase flows,” *Flow Measurement and Instrumentation,* pp. Volume 45, Pages 255-273, 2015.

[24] F. Scarano, “Tomographic PIV: principles and practise,” *Measurement Science and Technology,* pp. Volume 24, Number 1, 2013.

[25] V. Bertola, “Modelling and Experimentation in Two-Phase Flow,” *International Centre for Mechanical Sciences (Courses and Lectures),* pp. Volume 450, Pages 281-323, 2003.

[26] “Signal Processing,” [Online]. Available: https://www.signal-processing.com/index.php.

[27] F. Lemoine and F. Grisch, Laser-Induced Fluorescence, in Laser Metrology in Fluid Mechanics, Hoboken: John Wiley & Sons, 2012.

[28] A. Charogiannis, J. S. An and C. Markides, “A simultaneous planar laser-induced fluorescence, particle image velocimetry and particle tracking velocimetry technique for the investigation of thin liquid-film flows,” *Experimental Thermal and Fluid Science,* pp. Volume 68, Pages 516-536, 2015.

[29] H. Kleine, “High-Speed Visualization of Compressible Flows - Potential and Limitations,” *29th International Symposium on Shock Waves 1,* pp. p27-34, 2015.

[30] U. Hampel, Woodhead Publishing Series in Electronic and Optical Materials, Woodhead Publishing, 2015.

[31] Z. Zhang, M. Bieberle, F. Barthel, L. Szalinski and U. Hampel, “Investigation of upward cocurrent gas-liquid pipe flow using ultrafast X-ray tomography and wire-mesh sensor,” *Flow Measurement and Instrumentation,* pp. Volume 32, Pages 111-118, 2013.

[32] F. Hernandez-Alvarado, S. Kleinbart, D.V. Kalaga, S. Banerjee, J.B. Joshia and M. Kawaji, “Comparison of void fraction measurements using different techniques in two-phase flow bubble column reactors,” *International Journal of Multiphase Flow,* 2018.

[33] C. Tibiriçá, F. J. d. Nascimento and G. Ribatski, “Film thickness measurement techniques applied to micro-scale two-phase flow systems,” *Experimental Thermal and Fluid Science,* pp. Volume 34, Issue 4, Pages 463-473, 2010.

[34] B. D. Pauw, S. Vanlanduit, K. V. Tichelen, T. Geernaert, K. Chah and F. Berghmans, “Benchmarking of deformation and vibration measurement techniques for nuclear fuel pins,” *Measurement,* pp. Volume 46, Issue 9, Pages 3647-3653, 2013.

[35] F. D'Auria, Thermal Hydraulics in Water-Cooled Nuclear Reactors, Woodhead Publishing, 2017.

[36] “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation,” NEA/CSNI/R(2016)4, 1993.

[37] “CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients,” NEA/CSNI/R(1996)17, 1996.

[38] “Validation Matrix for the Assessment of Thermal-Hydraulic Codes for VVER LOCA and Transients,” NEA/CSNI/R(2001)4, 2001.

[39] “Containment Code Validation Matrix,” NEA/CSNI/R(2014)/3, 2014.

[40] R. Fortman, “Welcome to Stern Laboratories - General Presentation,” Stern Laboratories Inc., 2018.

[41] “Stern Laboratories Brochure,” Stern Laboratories Inc..

[42] “AREVA’s Thermo-Hydraulic Platform, Focus on Technology,” AREVA Brochure http://de.areva.com/customer/liblocal/docs/KUNDENPORTAL/Sonstige%20Broschüren/Thermo-hydraulic%20Plattform_10_2017.pdf, 2017.

[43] I. Ganzmann, W. Herr, H. Schmidt, W. Stecher, D. Walter, K. Umminger, A. Beisiegel, M. Wich, P. Dolleans and T. Muller, “AREVA's accredited testing facility,” *VGB PowerTech,* pp. 58 - 63, 1-2/2011.

[44] S. Leyer and M. Wich, “The Integral Test Facility Karlstein,” *Science and Technology of Nuclear Installations,* vol. 2012, Article ID 439374.

[45] K. Umminger, L. Dennhardt, S. Schollenberger and B. Schoen, “Integral Test Facility PKL: Experimental PWR Accident Investigation,” *Science and Technology of Nuclear Installations,* vol. 2012, Article ID 891056.

[46] H.-M. Prasser, M. Beyer, H. Carl, S. Gregor, A. Manera, H. Pietruske, P. Schuetz and F.-P. Weiss, “TOPFLOW: Design, Results and Perspectives,” Annual Report 2004, HZDR-420, February 2005.

[47] T. Seidel, D. Lucas and M. Beyer, “TOPFLOW Experiments on Direct Condensation and Bubble Entrainment,” HZDR-066, January 2016.

[48] S. Kliem, H.-M. Prasser, T. Sühnel, F.-P. Weiss and A. Hansen, “Experimental determination of the boron concentration distribution in the primary circuit of a PWR after a postulated cold leg small break loss-of-coolant-accident with cold leg safety injection,” *Nuclear Engineering and Design,* vol. 238, no. 7, pp. 1788-1801, July 2008.

[49] C. Vallée, D. Lucas, M. Beyer, H. Pietruske, P. Schütz and H. Carl, “Experimental CFD Grade Data for Stratified Two-phase flows,” *Nuclear Engineering and Design,* vol. 240, no. 9, pp. 2347-2356, September 2010.

[50] U. Hampel, “Directory of experimental facilities used for the Thermal-Hydraulics of Innovative Nuclear Systems (THINS) Project,” EURATOM grant agreement no. 249337, (2007-2011), https://www.ifrt.kit.edu/thins/, November 2013.

[51] M. Kumar, A. Dasgupta, A. Vishnoi, V. Jain, N. Kumar, K.Bodkha, D. Chandraker, A. Nayak, A. Pal and P. Vijayan, “Scaling Philosophy and System Description of AHWR Thermal-Hydraulic Test Facility (ATTF),” BARC/2010/E/007, 2010.

[52] V. Jain, “Indian Project on Multi-physics and Multi-scale Integrated Simulation of Nuclear Reactors (PROMISIN),” in *IAEA Workshop on Advanced Code Suites for Design, Safety Analysis and Operation of Heavy Water Reactors*, Ottawa, Canada, October 2012.

[53] A. Nayak, “Reliability Analysis of Passive Systems”.

[54] P. Dubey, G. Rao, D. Pilhawal, P. Vijayan and D. Saha, “Analysis of Experimental Data on Two-phase Natural Circulation from the Flow Pattern Transition Instability Studies Facility at APSARA Reactor,” BARC Report BARC/2004/E/031, 2004.

[55] “Natural Circulation Phenomena and Modelling for Advanced Water Cooled Reactors,” IAEA TECDOC 1677, 2012.

[56] R. Ferri, “The SIET Company - Areas of Business, Facilities, Capabilities,” Presented December 2016.

[57] M. Rigamonti and R. Silverii, “SIET Experimental Test Facilities for the ALWR Projects,” *Energia Nucleare,* vol. 13, no. 1, pp. 33 - 41, April 1996.

[58] R. Ferri, A. Achilli, G. Cattadori, F. Bianchi and P. Meloni, “Design Experiments and RELAP5 Code Calculations for the PERSEO Facility,” *Nuclear Engineering and Design,* vol. 235, pp. 1201 - 1214, 2005.

[59] C. Medich, M. Rigamonti and M. Tarantini, “SPES-2 The Experimental Test Facility Simulating the AP600 Plant,” *Energia Nucleare,* vol. 13, no. 1, pp. 42 - 53, April 1996.

[60] “Research Needs of Thermal-Hydraulic Tests for BWR,” Hitachi-GE Nuclear Energy Ltd, AE-OG-5931, 16 June 2017.

[61] M. Rohde and J. March-Leuba, “LAPUR 6.0 Benchmark Against Data from the GENESIS Facility,” US NRC NUREG/CR-7047, March 2012.

[62] C. T'Joen and M. Rohde, “Experimental study on the stability of a natural circulation driven super-critical water cooled reactor,” in *22nd International Symposium on Transport Phenomena*, Delft, The Netherlands, November 2011.

[63] G. Mignot, J. Dupont, D. Paladino and H.-M. Prasser, “LINX: A Separate Test Facility for Containment Safety Studies,” in *NUTHOS-11: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety*, Gyeongju, Korea, October 2016.

[64] D. Paladino and J. Dreier, “PANDA: A Multipurpose Integral Test Facility for LWR Safety Investigations,” *Science and Technology of Nuclear Installations,* Vols. 2012, Article ID 239319.

[65] “The NEA CFD Benchmark Exercise,” NEA/CSNI/R(2016)2, February 2016.

[66] D. Suckow and S. Güntay, “The DRAGON aerosol research facility to study aerosol behaviour for reactor safety applications,” in *International Conference on Reactor Physics, Nuclear Power: A Sustainable Resource*, Interlaken, Switzerland, September 2008.

[67] P. Ferroni, “Westinghouse's main thermal-hydraulic facilities and testing,” in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[68] “Westinghouse Hot Cell Facility and Laboratories brochure,” Westinghouse Electric Company LLC, 2017.

[69] C. Copenhafer, R. Lu, M. Conner and L. Smith, “Advanced Thermal Hydraulic Test Methods for Commercial Nuclear Design,” in *The 7th International Youth in Nuclear Congress (IYNC)*, Charlotte, North Carolina, August 2012.

[70] S. Anderson, A. Mingo and M. Majed, “Westinghouse BWR and PWR Fuel Thermal Hydraulic Test Facility in Sweden,” in *Nuclear Energy for New Europe 2008*, Portorož, Slovenia, September 2008.

[71] R. Vaghetto, “Texas A&M University Thermal-Hydraulic Laboratory - Overview and Capabilities,” in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[72] “Experimental facilities for gas-cooled reactor safety studies,” NEA/CSNI/R(2009)8, https://www.oecd-nea.org/nsd/docs/2009/csni-r2009-8.pdf, 2009.

[73] Z. Zhang, Y. Dong, F. Li, Z. Zhang, H. Wang, X. Huang, H. Li, B. Liu, X. Wu, H. Wang, X. Diao, H. Zhang and J. Wang, “The Shandong Shidao Bay 200 MWe High-Temperature Gas-Cooled Reactor Pebble-Bed Module (HTR-PM) Demonstration Power Plant: An Engineering and Technological Innovation,” *Engineering,* vol. 2, no. 1, pp. 112-118, March 2016.

[74] Z. Liu, C. Ren, Y. Sun, X. Yang and S. Jiang, “Introduction to the Chinese HTR-PM pebble bed equivalent conductivity test facility,” in *HTR2014: 7th International Topical Meeting on High Temperature Reactor Technology*, Weihai, China, July 2014.

[75] Y. Zhou, P. Hao, F. Li, L. Shi, F. He, Y. Dong and Z. Zhang, “Experiment study on thermal mixing performance of HTR-PM reactor outlet,” *Nuclear Engineering and Design,* vol. 306, pp. 186-191, 2016.

[76] D. Lisowski, C. Gerardi and C. Grandy, “Argonne Nuclear Engineering: Thermal Hydraulic Testing Capabilities,” in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, July 2017.

[77] D. Lisowski, C. Gerardi, D. Kilsdonk, N. Bremer, S. Lomperski, R. Hu, A. Kraus, M. Bucknor, Q. Lv, T. Lee and M. Farmer, “Final Project Report on RCCS Testing with the Air-based Natural convection Shutdown heat removal Test Facility (NSTF),” Nuclear Engineering Division, ANL-ART-47, August 2016.

[78] S. Lomperski, C. Gerardi and W. Pointer, “Fiber optic distributed temperature sensor mapping of a jet-mixing flow field,” *Experiments in Fluids,* vol. 56, no. 55, March 2015.

[79] “Experimental Facilities in Support of Liquid Metal Cooled Fast Neutron Systems,” IAEA Nuclear Energy Series, NP-T-1.15, October 2018.

[80] O. Gastaldi, G. Rodriguez, L. Ayrault, I. Tkaschenko, B. Collard, E. Sanseigne, J. Dumesnil, F. Dujet, F. Serre and G. Willermoz, “Experimental platforms in support of the ASTRID program: Existing and planned facilities at CEA,” in *Technical Meeting on Existing and Proposed Experimental Facilities for Fast Neutron Systems*, Vienna, June 2013.

[81] D. Guénadou, I. Tkatshenko and P. Aubert, “PLATEAU Facility in Support to ASTRID and the SFR Program: An overview of the first mock-up of the ASTRID upper plenum, MICAS,” in *NURETH-16*, Chicago, September 2015.

[82] “Natural circulation in water cooled nuclear power plants: Phenomena, models, and methodology for system reliability assessments,” IAEA-TECDOC-1474, November 2005.

[83] A. Onea, W. Hering, J. Reiser, A. Weisenburger, N. Diez de los Rios Ramos, M. Lux, R. Ziegler, S. Baumgärtner and R. Stieglitz, “Development of high temperature liquid metal test facilities for qualification of materials and investigations of thermoelectrical modules,” *Materials Science and Engineering,* vol. 228, July 2017.

[84] J. Pacio, M. Daubner, T. Wetzel, I. Di Piazza, M. Tarantino, D. Martelli and M. Angelucci, “Experimental Nusselt Number in Rod Bundles Cooled by Heavy-Liquid Metals,” in *26th International Conference on Nuclear Engineering, ICONE26*, London, July 2018.

[85] “Experimental Installations book,” H2FC European Infrastructure Project, FP7 Grant Agreement No. FP7-284522, April 2012.

[86] "Multi-purpose hYbrid Research Reactor for High-tech Applications: A research infrastructure for a new era," SCK-CEN, MYRRHA Technical Brochure, November 2016.

[87] S. Jérôme, M. Allibert, O. Beneš, S. Delpech, O. Feynberg, V. Ghetta and D. Heuer, "The molten salt reactor (MSR) in generation IV: Overview and perspectives," *Progress in Nuclear Energy,* vol. 77, pp. 308-319, 2014.

[88] K. Furukawa, Y. Kato and C. W. Urban, "Thorium Molten Salt Nuclear Energy Synergetics System for the Global Energy Demand".

[89] Y. Ritsuo, K. Mitachi and M. Kinoshita, "Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES)," in *Molten salts in nuclear technology*, 2013.

[90] P. R. Rubiolo, R. M. T, V. Ghetta and J. Giraud, "High temperature thermal hydraulics modelling of a Molten Salt: Application to a Molten Salt Fast Reactor (MSFR)," *ESAIM: Proceedings and Surveys,* vol. 58, pp. 98-117, December 2017.

[91] J. E. O'Brien, S.-J. Yoon, P. Sabharwall and S. Bragg-Sitton, "High-Pressure, High-Temperature Thermal Hydraulic Facility for Nuclear-Renewable Hybrid Energy System Studies; Facility Design Description and Status Report," INL, INL/EXT-17-43269, September 2017.

[92] P. Sabharwall, C. Stoots, D. M. McEligot, R. Skifton and H. M. McIlroy, "Matched-Index-of-Refraction Flow Facility for Fundamental and Applied Research," INL, INL/CON-14-32432, November 2014.

[93] L. Leung, "Achievements of Phase-I Thermalhydraulics and Safety Program in Support of Canadian SCWR Concept Development," in *The 6th International Symposium on Supercritical Water-Cooled Reactors, ISSCWR6-13003*, China, March 2013.

[94] "Thermal Hydraulics & Severe Accident research Division - Annual Report 2017," THSARD 2017, KAERI/GP-529/2018, December 2018.

[95] "Report of the OECD/NEA KAERI Rod Bundle CFD Benchmark Exercise," NEA/CSNI/R(2013)5, July 2013.

[96] A. Morozov, "Experimental Investigation of Operation of VVER Steam Generator in Condensation Mode in the Event of the Severe Accident," in *Proceedings of the International Youth Nuclear Congress (IYNC)*, 2008.

[97] "Experimental Facilities at KTH/CEKERT Relevant to Nuclear Engineering Research," https://www.cekert.kth.se/polopoly_fs/1.547462!/Nuclear%20Research%20Facilities%20at%20KTH.pdf.

[98] "Report on the OECD-NEA Vattenfall T-Junction Benchmark exercise," NEA/CSNI/R(2011)5, May 2011.

[99] B. Phillips, "Thermal-hydraulics Experimental Facilities at MIT NSE," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[100] L. A. Carrilho, "Experimental and computational study of roughened surface for PWR rod bundles," University of South Carolina, PhD Dissertation, 2012.

[101] P. Sabharwall, "Thermal-Hydraulic Experimental Facilities for Advanced Reactor Development at INL," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[102] M. Anderson, "UW-Madison Thermal Hydraulics Laboratory Facilities," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[103] W. Marcum, "Light Water Reactor Test Facilities - Integral and Separate Effects," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[104] R. Adams, "High-resolution Gamma Tomography at the University of Michigan," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[105] S. Kim and I. Mamoru, "Purdue University Thermal Hydraulic Experiment Capability LWR & SMR Group," in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[106] J. Buell, D. Dormuth, P. Ingham and R. Swartz, "RD-14M Facility Description," in *Presented to US Nuclear Regulatory Commission*, June 2003.

[107] H. Glaeser and H. Karwat, “The contribution of UPTF experiments to resolve some scale-up uncertainties in countercurrent two phase flow,” *Nuclear Engineering and Design,* vol. 145, pp. 63-84, 1993.

[108] “SBWR Testing Summary Report,” NEDO-32606, GE Nuclear Energy Project 717 for USNRC, August 2002.

[109] R. Ferri, “Simulating capabilities of the SPES3 facility,” SIET ENEA Report RdS/2011/6, September 2011.

[110] J. Miller and B. Martin, “BWXT Integrated System Test Facility,” NSUF/GAIN Nuclear Thermal-Hydraulics Workshop, Idaho Falls, 13 July 2017.

[111] J. Woodcock and M. B. Dzodzo, “Application of Large Scale Containment Database to AP600 Loss-Of-Coolant-Accident Internal Circulation and Stratification Evaluations,” in *ICONE-8488, ICONE 8 Conference*, April 2000.

[112] T. Sevon, “MELCOR Modeling of a Passive Containment Cooling System Experiment PANDA T1.1,” VTT Energy VTT-R-01433-12, 2012.

[113] S.-D. Hong, C.-S. Kim, J.-H. Kim and M.-H. Kim, “Status of VHTR-related Thermal-Hydraulic Experimental Research at KAERI,” in *19th Pacific Basin Nuclear Conference (PBNC)*, Canada, 2014.

[114] X. Sun, “UM Thermal Hydraulics Laboratory: Overview and Capabilities (HTGR and MSR),” in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[115] P. R. Rubiolo, V. Ghetta, J. Giraud and M. T. Retamales, “High temperature thermal hydraulics modelling of a Molten Salt: Application to a MSFR,” in *Workshop Écoulements à Bas Mach*, Orsay, November 2015.

[116] A. k. Srivastava, A. Borgohain, S. S. Jana, R. K. Bagul, R. R. Singh, N. K. Maheshwari, D. G. Belokar and P. K. Vijayan, “Experimental and Theoretical Studies in Molten Salt Natural Circulation Loop (MSNCL),” BARC/2014/E/016, 2014.

[117] A. K. Srivastava, A. Borgohain, N. K. Maheshwari and P. K. Vijayan, “Experimental Studies on Natural Circulation in Molten Salt Loops,” in *Thorium Energy Conference 2015 (ThEC15)*, October 2015.

[118] A. Cammi, M. Cauzzi, L. Luzzi and A. Pini, “DYNASTY: An Experimental Loop for the Study of Natural Circulation with internally Heated Fluids,” in *12th International Conference on Heat Transfer, Fluid Mechanics and Thermodynamics (HEFAT2016)*, July 2016.

[119] A. Sagara, “R&D Activities on flibe blanket in Japan,” in *ITER TBM Project Meeting at UCLA*, February 2004.

[120] G. Yoder, “Liquid Salt Experimentation to Support FHR and MSR Development,” in *NSUF/GAIN Nuclear Thermal-Hydraulics Workshop*, Idaho Falls, 13 July 2017.

[121] N. Zweibaum, J. Bickel, Z. Guo, L. Huddar, J. Kendrick, M. Laufer and P. Peterson, “UC Berkeley Compact Integral Effects Test (CIET): Facility Design, Test Program, and Initial Validation Studies,” in *Workshop on Molten Salt Reactor Technologies, ORNL*, October 2015.

[122] P. Ferroni et al, “Westinghouse’s Lead Fast Reactor Program,” in *American Nuclear Society Winter Meeting & Expo 2017*, Washington, November 2017.

ANNEX A - THERMAL HYDRAULIC TEST FACILITIES

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp °C (pressure MPa)
Heat transfer test facility, uOttawa, Canada [93]	-	CO2, SCWR	2011-	Supercritical heat transfer	- (10.0)
Natural Circulation & Stability facility, Manitoba, Canada [93]	5	CO2, SCWR	2012-	Supercritical natural circulation & stability	95 (12.0)
Supercritical Refrigerant Test Loop, Carleton, Canada [93]	-	R-134a, SCWR	2012-	Supercritical heat transfer	140 (6.0)
Supercritical critical flow test facility, Montreal, Canada [93]	-	water, SCWR	2013-	Supercritical critical flow	625 (26.0)
Critical Heat Flux tests, Stern Laboratories, Canada [40]	16000	CANDU, PWR, BWR	In operation	CHF, BT and DNB tests	340 (18.0)
ROMÉO & JULIETTE, Framatome, France [42]	-	Water, PWR	In operation	RPV flow distribution	100 (1.6)
MAGALY, Framatome, France [42]	-	Water, PWR	In operation	FIV of control rod cluster	100 (1.6)
KATHY (KArlstein Thermal HYdraulic test loop), Framatome, Germany [42]	20000	Water, PWR	In operation	Full scale CHF	360 (18.5)
KOPRA, Framatome, Germany [42]	1000	BWR, PWR,	1986-	Multifunction component tests	360 (19.4)
BENSON, Framatome, Germany [42]	2000	Water, PWR	In operation	HP heat transfer, CHF	600 (33)
PETER (PWR fuel Element Tests at ERlangen), Framatome, Germany [42]	2000	PWR, BWR	In operation	Full scale fuel FIV	- (-)
APPEL (AREVA Pump Test Loop), Framatome, Germany [42]	450	Water, PWR	In operation	pump test loop	200 (10)
GAP, Framatome, Germany [42]	-	Water, PWR	In operation	large valve test facility	350 (16.5)
COSMOS-L (Critical-heat-flux On Smooth and MOdified Surfaces - Low pressure loop), KIT, Germany[43]	75	Water, PWR	-	Convection boiling, CHF, impact of surface on boiling process	- (0.2)
COSMOS-H (High pressure loop), KIT, Germany[32]	1200	Water, PWR	In construction	Convection boiling, CHF	360 (17)
WENKA (Water ENtrainment Channel KArlsruhe), KIT, Germany[32]	-	Water, PWR	2001-	Counter-current stratified two-phase flows	- (-)

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[43] http://www.iket.kit.edu/english/index.php

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp ^{o}C (pressure MPa)
HAWAC (Horizontal Air/WAter Channel) HZDR, Germany [49]	-	Air, Water, PWR	In operation	Stratified two-phase flows	20 (0.1)
ROCOM (ROssendorf COolant Mixing model), HZDR, Germany [48]	-	Water, PWR	In operation	Coolant mixing	20 (0.1)
TOPFLOW (Transient twO Phase FLOW) Test Facility, HZDR, Germany [46]	4000	Water, BWR	2003-	Two-phase flow phenomena	286 (7.0)
TOPFLOW+, HZDR, Germany	-	Water, air, PWR	2014-	Flow structures and phenomena in valves / obstacles	- (0.1)
SCUBA, TUM, Germany[44]	-	Water, air, PWR, BWR	In operation	Steam bubble condensation	98 (0.14)
Apsara natural circulation loop, BARC, India [54]	10	Water, AHWR	In operation	Two-phase natural circulation	285 (7.0)
Boiling Water Loop, BARC, India	2400	Water, AHWR	In operation	Two-phase flow, CHF, pressure drop	310 (9.8)
FISBE (Facility for Integral System Behaviour Experiments), BARC, India	3000	Water, AHWR	In operation	Two-phase natural circulation	343 ()
HPNCL (High Pressure Natural Circulation Loop), BARC, India [53]	80	Water, AHWR	In operation	Two-phase natural circulation	315 (11.2)
PCL (Parallel Channel Loop), BARC, India [53]	200	Water, AHWR	In operation	Natural circulation	220 (2.0)
GEST (Generatore Esperienze Sicurezza Termoidrauliche), SIET, Italy [56]	10000	Water, PWR	1984-	Steam generators, separators and dryers tests	- (17)
IETI (Impianto per Esperienze Termo-Idrauliche), SIET, Italy [56]	10000	Water, PWR	1992-	Power channel flow, steam jet pumps	400 (22.0)
PANTHERS (Performance ANalysis and Testing of HEat Removal Systems), SIET, Italy [57]	20000	Water, ESBWR	1994-	PCCS and ICS performance tests	310 (10.0)
PERSEO (in-Pool Energy Removal System for	20000	Water, PWR	2002-	IC heat exchanger performance	310 (10.0)

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[44] https://www.ntech.mw.tum.de/index.php?id=113&L=1

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp ^{o}C (pressure MPa)
Emergency Operation), SIET, Italy [58]					
HICOF (HItachi COre and Fuel thermal hydraulic test loop), Hitachi-GE, Japan [60]	4000	Water, BWR	1984-1989	Transient boiling transition, critical power, axial pressure drop	290 (7.0)
FIVE (Flow Induced Vibration Experiment loop), Hitachi-GE, Japan [60]	1000	Water, BWR	2007-2008	Acceleration, strain of tube, FIV	288 (7.2)
HUSTLE (Hitachi Utility Steam Test Leading facility), Hitachi-GE, Japan [60]	1000	Water, BWR	2009-	Void fraction, axial pressure drop, pressure fluctuation	290 (7.0)
STNS (Steam Test facility for Nuclear Safety systems), Hitachi-GE, Japan [60]	-	Water, BWR	-	Heat exchanger, tube temperature, void fraction in pool	245 (3.5)
ATHER (Advanced Thermal-Hydraulic Evaluation of Reflood Phenomena), KAERI, Korea [94]	-	Water, PWR	In operation	Reflood heat transfer, vapour heat transfer	- (0.6)
B&C (Blowdown and Condensation), KAERI, Korea [94]	-	Water, PWR	In operation	Critical flow, air clearing, steam condensation, and thermal mixing tests	- (16.1)
FINCLS (Facility to Investigate Natural Circulation in SMART), KAERI, Korea [94]	60	Water, PWR	2017-	Single and two-phase natural circulation	360 (17.0)
FTHEL (Freon Thermal Hydraulic Experimental Loop), KAERI, Korea [94]	720	R-134a, PWR	In operation	Thermal mixing and CHF for rod bundles	150 (4.5)
MATiS-H (Measurement and Analysis of Turbulent Mixing in Subchannels – Horizontal), KAERI, Korea [95]	-	Water, PWR	In operation	Cold test loop for rod bundles	- (0.1)
MIDAS (Multi-dimensional Investigation in Downcomer Annulus Simulation), KAERI, Korea [94]	-	Water, PWR	In operation	Steam-water phenomena in the downcomer annulus	300 (1.0)
RCPTF (Reactor Coolant Pump Test Facility), KAERI, Korea [94]	10400	Water, APR1400	2012-	RCP performance	350 (17.0)

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp °C (pressure MPa)
RCS (Reactor Coolant System Heat Transfer Test Facility), KAERI, Korea [94]	970	Water, PWR	In operation	CHF, post-CHF and reflood heat transfer	346.85 (16.0)
SUBO (SUBcOoled Boiling) test facility, KAERI, Korea [94]	90	Water, PWR	In operation	Subcooled boiling phenomena	- (0.155)
VAPER (VAlve Performance Evaluation test Rig), KAERI, Korea [94]	-	Water, PWR	In operation	Valve performance for Safety Injection Tank	- (4.5)
GENESIS, TU Delft, Netherlands [61]	25	Freon 134a, ESBWR	2005-	Boiling, natural circulation facility	44 (1.1)
DeLight (Delft Light water reactor facility), TU Delft, Netherlands [62]	18	Freon R23, SCWR	2009-	Natural circulation stability	33 (5.7)
SEEDS (SEven rods bundle Experiments in Delft for SESAME), TU Delft, Netherlands	-	Water	2017-	FIV, Flow phenomena in rod bundles	- (-)
HA2M-SG, IPPE, Russia [96]	1000	Water, VVER	In operation	Passive safety system, condensation	200 (1.6)
HWAT (High-pressure WAter Test) loop, KTH, Sweden [97]	1000	Water, PWR	In operation	Singe, two-phase flow, CHF	- (25.0)
LFAT (Low-pressure Fuel Assembly Test) loop, KTH, Sweden [97]	15	Water, PWR	In operation	Atmospheric pressure, adiabatic two-phase flow	- (0.1)
THEMFE (Thermal Mixing and Fatigue) loop, KTH, Sweden [97]	-	Water, PWR	In operation	Temp fluctuations in solid walls	- (-)
T-Junction, Vattenfall, Sweden [98]	-	Water, PWR	2008-	T-Junction hot and cold mixing	- (-)
FRIGG, WEC, Sweden [67]	15000	Water, BWR	In operation	Dryout (incl. instability), DP, hydraulic stability	311 (10)
ODEN Loop, WEC, Sweden [67]	12000	Water, PWR	In operation	Heat transfer, DNB, DP, TDC	366 (20.0)
BURE, WEC, Sweden [67]	-[45]	Water, BWR	In operation	FIV, Fretting	300 (8)
FRODE, WEC, Sweden [67]	-[4]	Water, BWR	In operation	Debris catching, DP, flow distribution, lift forces	80 (0.1)

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[45] Hydraulic loop, so maximum power is not relevant

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp ^{o}C (pressure MPa)
AROBE (Adiabatic ROd Bundle Experiments), PSI, Switzerland[46]	-	Water, air	In operation	Two-phase flow with neutron imaging	- (-)
CNHT facility, PSI, Switzerland[19]	-	Water	In operation	Fuel bundle boiling and dryout with cold-neutron and X-ray imaging	- (-)
FLORIS, PSI, Switzerland[19]	-	Water	In operation	Density driven flow phenomena in BWR lower plenum	- (-)
GEMIX (GEneral MIXing Experiment), PSI, Switzerland[19]	-	Water	In operation	Turbulent and laminar plane wake mixing	- (-)
LINX (Large scale Investigation of Natural circulation and miXing), PSI, Switzerland [63]	-	Water	1999-	Containment safety, natural circulation, mixing and condensation	250 (1.0)
SUBFLOW (SUB-channel FLOW test facility), PSI, Switzerland[19]	-	Water, air	In operation	Fuel rod bundle flow, two-phase flow	- (-)
T-Junction, PSI, Switzerland[19]	-	Water	In operation	Mixing of hot and cold streams	- (-)
FACTS (Fuel Assembly Compatibility Test System) Loop, WEC, USA [67]	-[4]	Water, PWR	In operation	DP, FIV, debris mitigation	120 (15.5)
VIPER (Vibration Investigation and Pressure drop Experimental Research) Loop, WEC, USA [67]	-[4]	Water, PWR	In operation	DP, FIV, Fretting	200 (2.5)
VISTA (Vibration Investigation of Small-scale Test Assemblies) loop, WEC, USA [67]	-[47]	Water, PWR	In operation	DP, FIV	20 (0.1)
WALT (Westinghouse Advanced Loop Tester), WEC, USA [67]	100	Water, PWR	2005-	Crud simulation, DNB, LOCA	345 (15.5)
WATCH (Westinghouse Annular Thermal Crud Hydraulic), WEC, USA [67]	90	Water, BWR	In operation	Crud simulation, Dryout, LOCA	345 (12)

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[46] https://www.psi.ch/teg/facilities

[47] Hydraulic loop, so maximum power is not relevant

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp °C (pressure MPa)
PEThER (Platform for Experimental ThERmal-hydraulics), MIT, USA [99]	3000	Water, BWR, PWR	In operation	Suppression of nucleate boiling in annular flow	180 (10)
SHELT (Single Heater Element Loop Tester), USC, USA [100]	10	Water, PWR	In operation	Convective heat transfer, friction factors	120 (0.5)
CHF test Facility, TAMU, USA [71]	0.5 for Freon [48]	Freon 11 (or similar), PWR	In preparation	CHF	204 (3.5) for Freon
Wire wrapped fuel assembly, TAMU, USA [71]	-	p-cymene, PWR	2015-?	Pressure drop, laminar, transition, & turbulent regimes	- (-)
Helical coil SG test loop, TAMU, USA [71]	-	Water, PWR	In operation	Pressure drop and velocity variation	- (-)
PTS (Pressurized Thermal Shock) Facility, TAMU, USA[49]	-	Water, PWR	In operation	Mixing phenomena in a horizontal cold leg	- (-)
ARTIST (Advanced Reactor Technology Integral System Test), INL, USA [101]	-	Helium, FLiNaK, water, PWR	Under design	Heat transfer, FIV, multipurpose and multi-fluid test facility	325 (15.0)
MIR (Matched Index of Refraction) facility, INL, USA [101][50]	8.5	Oil, PWR	2001-	Entropy generation rate in boundary layer	26 ()
Heat transfer and CHF, UW-M, USA [102]	400	Water, PWR	In operation	Heat transfer, CHF, flow oscillations	550 (25.0)
TRTL (Transient Research Test Loop), OSU, USA [103]	500	Water, PWR	2015-	Steady and transient CHF loop	603 (20.0)
Hydro-Mechanical Fuel Test Facility, OSU, USA [103]	-	Water, BWR, PWR	2011-	Bundle flow test loop, FSI validation	- (-)
PCHT (Post-CHF Heat Transfer), UM, USA [104]	-	Water, PWR	In operation	Post-CHF, flow regime transition	- (3.4)
Small diameter Pipe loop, Purdue, USA [105]	-	Water, PWR	In operation	Two-phase flow data	- (-)
Annular flow loop, Purdue, USA [105]	-	Freon-113, water, BWR	In operation	Droplet, entrainment and film dynamics	- (1.0)

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[48] Corresponds to 5000 kWe power and 18 MPa for water and PWR conditions
[49] http://thrlab.tamu.edu/category/current-projects
[50] https://mir.inl.gov

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, status	Test type	Max operating temp ^{o}C (pressure MPa)
HP boiling loop, Purdue, USA [105]	-	Water, PWR	In operation	Boiling, condensation, CHF	- (1.0)
Fuel assembly simulation loop, Purdue, USA [105]	-	Water, BWR	In operation	Two-phase flow in rod bundle, spacer grid effects	- (1.0)
Natural Circulation SMR loop, Purdue, USA [105]	18	Water, BWR NMR-50	In operation	Natural circulation start-up instability	- (1.0)
Pressurized SMR loop, Purdue, USA [105]	18	Water, PWR SMR	In operation	Blowdown, DBA, ECCS performance	- (1.0)

Table 3: SET Water Test Facilities

Facility name, Organisation, Country of origin	Max Power (MWe)	Reference reactor	Status, start date	Height scale; volumetric scale (pressure MPa)
CAPCN, PTC-CNEA, Argentina [55]	0.3	CAREM	-	1;280 (12.0)
RD-14M, CNL, Canada [106]	-	CANDU 6, ACR-1000	1987-	-;- (12.5)
PWR PACTEL (PArallel Channel TEst Loop), VTT, Finland [107]	1[51]	EPR Like-4L	2009-?	1;400 (8.0)
CLOTAIRE, CEA, France [82]	-	ESBWR	-	-;- (0.9)
INKA (INtegral Test Stand KArlstein), Framatome, Germany [42]	22	BWR	-	1;24 (16.0)
PKL-III (PrimärKreisLauf), Framatome, Germany [45]	2.5[34]	PWR-4L	1985-	1;145 (4.5)
ATTF (AHWR Thermal hydraulic Test Facility), BARC, India [51]	9	AHWR	2013	1;226 (7.0)
ITL (Integral Test Loop), BARC, India [55]	3	AHWR	2005	1;452 (10)
SPES-2 (Simulatore PWR per Esperienze di Sicurezza), SIET, Italy [59]	9	AP600-2L	1994-?	1;430 (20.0)
ROSA-AP600 (LSTF), JAERI, Japan[52] [55]	10[34]	AP600-2L	1995-?	1;30.5 (16.0)
GIRAFFE (Gravity-driven Integral Full Height Test for Passive Heat Removal), TOSHIBA, Japan [108]	-	ESBWR	1996-	1;400 (-)
SNUF (SNU Facility), SNU, Korea[35]	0.27	APR1400-2L	-	0.16;1139 (0.8)
ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation), KAERI, Korea [94]	1.96[34]	APR1400-2L	2007-	0.5;288 (20.0)
FESTA (Facility for Experimental Simulation of Transients and Accidents), KAERI, Korea [94]	3	SMART-IWCR	2012-	1;49 (18.0)
VISTA-ITL (Verification by Integral Simulation of Transients and Accidents), KAERI, Korea [94]	0.819	SMART-IWCR	2009-	0.36;1310 (17.2)
PANDA (Passive Nachwärmeabfuhr- und Druckabbau-Testanlage), PSI, Switzerland [64]	1.5	SBWR, ESBWR	1991-	1;40 (1.0)
GIST (Gravity-driven Integrated Systems Test), GE, USA [108]	-	ESBWR	1988-?	1;508 (-)
PUMA (Purdue University Multi-Dimensional Test Assembly), Purdue, USA [55] [105]	0.65	SBWR, ESBWR	-	1;400 (-)
BWXT-IST, CAER, USA [109] [110]	1.8	mPower IWCR SMR	2012-	1;- 343 (17.2)
APEX (Advanced Plant EXperiment), OSU, USA[25]	0.6	AP600-2L	1994-?	0.25;192 (2.76)
APEX-AP1000, OSU, USA [107]	1[34]	AP1000		0.25;192 (2.76)
APEX-CE, OSU, USA[35]	0.65	CE-PWR-2L	-	0.29;276 (2.76)
OSU-MASLWR, OSU, USA[35] [55]	0.6	MASLWR-IWCR	2002-?	0.33;254 (11.4)
NIST-1 (NuScale Integral System Test), OSU, USA [109]	-	NuScale	2003-	0.33;- (-)
SRI-2, SRI, USA[35]	0.088	B&W PWR	-	0.25;255 (0.7)

Table 4: IET Water Test Facilities

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[51] Scaled full power was not possible [99]

[52] https://www.oecd-nea.org/tiethysweb

Facility name, Organisation, Country of origin	Coolant	Start date, status	Test type	Volume m^3 (pressure MPa)
Lucas heights blowdown/ containment, AAEC, Australia	Water	1981-?	Blowdown, condensation HTC	1.81 (-)
LSGMF (Large Scale Gas Mixing Facility), AECL, Canada	Helium	1997-	Stratification, mixing in gases	1000 (0.1)
LSVCTF (Large Scale Vented Combustion Test Facility) AECL, Canada	Hydrogen	1996-	Deflagration, jet ignition	120 (0.1)
SSBT (Small Scale Burst Test) facility), AECL, Canada	Water	2004-2012	Pressure wave propagation	- (18.0)
Bubble condenser facility, SVUSS, Czech Republic	Water	1997-	TH loads, bubble Condenser	21 (0.23)
MISTRA (Mitigation and STRAtification facility), CEA, France	Helium, Air, Steam	2002-2008	Convection HTC, Condensation, mixing in gases	100 (0.6)
SULTAN, CEA, France	Water	1995-	Boiling HTC, CHF	- (1.0)
ENACCEF (ENceinte d'ACCElération de Flamme), CNRS, France	Hydrogen, air	2007-	Deflagration, hydrogen flame acceleration, quenching	- (0.1)
TOSQAN (Test Station for Simulation and Qualification in Airborne Conditions), IRSN, France	Steam, air, helium	2002-2013	Convection HTC, Condensation, mixing in gases, mixing by sprays	7 (0.7)
HYKA (Hydrogen test centre), KIT, Germany [85]	Hydrogen	2004-	Deflagration, hydrogen flame acceleration, detonation	98 (10.0)
HDR (HeissDampfReaktor), KFK, Germany [111]	Water	1976-1991	Stratification, flashing, convection HTC, condensation, mixing in gases, effects of internal and external sprays, positions of releases, natural circulation	11300 (11.0)
BMC (Battelle Model Containment), BATTELLE, Germany [111]	Hydrogen steam	1986-1992	Deflagration, Hydrogen flame acceleration, natural circulation, stratification, positions of steam releases, positions and strengths of heat sources	640 (0.5)
THAI (Thermal-hydraulics, Hydrogen, Aerosols and Iodine), BT, Germany	Hydrogen iodine and aerosol	2000-2014	Convection HTC, mixing of gases, condensation, evaporation, stratification, deflagration	60 (1.4)
PSS (Pressure Suppression System), GKSS, Germany	Water	1979-?	Conduction, convection HTC, condensation, evaporation	240 (0.6)
GKM (GrossKraftwerk Mannhein) I & II, KWU, Germany	Water	1976-?	Steam blowdown pressure loads, condensation	- (0.28)

Facility name, Organisation, Country of origin	Coolant	Start date, status	Test type	Volume m^3 (pressure MPa)
CONAN (CONdensation with Aerosols & Noncondensable gases), UNIPI, Italy	Steam, air	2000-	Condensation on surfaces	- (-)
Full Scale Mk II CRT (Containment Response Test), JAERI, Japan	Water	1977-?	LOCA hydrodynamic loads for Mark II PSS	329 (7.0)
NUPEC model containment facility, NUPEC, Japan [111]	Helium	1993-	Conduction, convection HTC, Condensation, mixing in gases, effects of internal sprays, positions of steam releases	1300 used (0.14)
TIGER (Toshiba Innovative Geminate Test Loop for Reactor Safety System), TOSHIBA, Japan [112]	Water,	2010-	Condensation, horizontal PCCS integral tests	- (-)
BC-V-213 (Bubble Condenser), EREC, Russia	Water	1999-2003	Conduction, convection HTC, condensation, evaporation, mixing	- (0.3)
MARVIKEN, STUDSVIK, Sweden	Water	1976-?	Flashing, evaporation, condensation, mixing, critical flow	1970 (4.6)
DRAGON (DiveRse purpose Aerosol GeneratiON), PSI, Switzerland [66]	Steam, nitrogen, air	1994-	Multi-purpose aerosol generation. Soluble, hygroscopic aerosol and insoluble aerosol particles	1.6 (0.5)
ISOLDE, PSI, Switzerland	Water, air, nitrogen	In operation	Bubble hydrodynamic, mass transfer from gas to liquid	165 (0.7)
TRISTAN (Tube Rupture In Steam generator multi-phase flow investigations), PSI, Switzerland	Water, air, nitrogen	In operation	Bubble hydrodynamic, mass transfer from gas to liquid	50 (0.7)
VEFITA (Venting Filter Assessment), PSI, Switzerland	Steam, nitrogen	In operation	Filtered containment system performance, iodine and aerosol retention	200 (1.0)
PANDA (Passive Nachwärmeabfuhr-und Druckabbau-Testanlage), PSI, Switzerland [64]	Water	1991-	Stratification, boiling HTC, convection HTC, condensation, evaporation, mixing in gases	515 (1.0)
CVTR (Carolinas–Virginia Tube Reactor), CVNP, USA [111]	Water	1960-1967	Containment pressure response, internal sprays, stratification, natural circulation and steam release	6428 (0.15)
CSTF (Containment Systems Test Facility), HEDL, USA	H_2-steam, He-steam	1982-1984	Condensation, mixing, stratification	850 (0.5)
PSTF (Pressure Suppression Test Facility, GE, USA	Water	1972-1974	Suppression pool hydrodynamic loads, dynamic effects	- (-)
4T (Temporary Tall Test Tank), GE, USA	Water	1974-?	Containment tests, condensation	- (-)

Facility name, Organisation, Country of origin	Coolant	Start date, status	Test type	Volume m^3 (pressure MPa)
FSTF (Full-Scale Test Facility), GE, USA	Water	1979-?	Pressure load, pressure alleviation, chugging phase	- (-)
CYBL (Cylindrical Boiling facility), SNL, USA	Water	1995-	Boiling, CHF, convection HTC, condensation, flashing	- (0.1)
ULPU, UCSB, USA	Water	1993-2002	Boiling, CHF, convection HTC, condensation, flashing	- (-)
PCCS LST (Large Scale Test) facility, WEC, USA [111]	Water	1991-1996	Pressurisation transient, heat transfer, condensation distributions, positions of steam releases	83.1 (0.69)

Table 5: Containment Test Facilities for Water Reactors [7] [39]

Facility name, Organisation Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
ETF-HT (Engineering Test Facility – Helium Technology), INET, China [73]	10000	Helium, HTR-PM	2011-	Steam generator, helium circulator and component verification	750 (7.0)
TF-PBEC (Test Facility - Pebble-Bed Equivalent Conductivity), INET, China [74]	900	Helium, HTR-PM	In operation	Equivalent conductivity of pebble-bed	1600 (30 Pa)
TF-PBF3D (Test Facility - Pebble-Bed Flow 3D), INET, China [73]	-	Air, HTR-PM	In operation	Flow and pressure drop through pebble-bed	20 (0.1)
Reactor outlet mixing test facility, INET, China [75]	-	Air, HTR-PM	In operation	Thermal mixing and pressure drop	70 (0.1)
ESTHAIR, CEA, France [72]	-	Air, ALLEGRO	-	Heat transfer / pressure drop through core	20 (0.1)
HECO (Helium loop), CEA, France [72]	-	Helium, ALLEGRO	2011-	Forced convection in decay heat removal loop	260 (7.0)
HEDYT (Helium Dynamic Test), CEA, France [72]	100	Helium, ALLEGRO	2009-	Dynamic helium tests in a loop	850 (10.0)
HELITE, CEA, France [72]	-	Helium, ALLEGRO	2012-	Qualification of thermal shielding and heat exchanger	850 (10.0)
SALSA, CEA, France [72]	180	Air, ALLEGRO	-	Natural and forced convection in primary loop and decay heat removal loops at 1/3 scale	200 (1.0)
L-STAR (Luft - STab, Abstandshalter, und Rauigkeiten), KIT, Germany [50]	24	Air, GFR, (V)HTR	2007-	Heat transfer and pressure drop in annular fuel channel	200 (0.4)
NACOK (Natural flow inside core with corrosion), FZJ, Germany [72]	160	Air	-	Graphite oxidation, thermal hydraulics and pebble bed heat transfer	1200 (-)
GP Loop (Gas Particle Loop), HZDR, Germany [50]	-	Air, (V)HTR	2010-	Deposition and resuspension of particles	20 (0.1)
HTTR (High Temperature engineering Test Reactor), JAEA, Japan[53] [72]	30000	Helium	2002-	Core and plant transient tests for HTGRs	950 (-)
HELP (Helium Experimental Loop), KAERI, Korea [113]	150	Helium, VHTR	2011-	Scaled tests of heat exchanger and fuel block	500 (9.0)

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[53] https://httr.jaea.go.jp/eng/

Facility name, Organisation Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
NACEF (NAtural Cooling Experimental Facility), KAERI, Korea [113]	10	Air, VHTR	2015-	Reactor Cavity Cooling System natural convection cooling	- (-)
HOMER-GAMILO (HOrizontal Mixing Experiment in a Rectangular channel), PSI, Switzerland [50]	-	Nitrogen, helium, (V)HTR	In operation	Turbulent mixing of different density gas streams	30 (0.1)
MAX fluid dynamics facility, ANL, USA [78]	-	Air	In operation	Jet mixing	70 (0.1)
NSTF (Natural Convection Shutdown Heat Removal Test Facility), ANL, USA [77]	220	Air, water, HTGR	2014-	Reactor Cavity Cooling System natural convection cooling	677 (0.1)
HTHF (High-Temperature Helium Test Facility), UM, USA [114]	46	Helium, HTGR	In operation	Test heat exchangers, valves and instrumentation	800 (3.0)
STL (S-CO_2 Test Loop), UM, USA [114]	-	CO_2, HTGR	In operation	Supercritical CO_2 test loop	630 (16.0)
Air-ingress Test Facility, UM, USA [114]	-	Air, HTGR	In operation	Air ingress into hot exit plenum	540 (0.35)
HTTF (High Temperature Test Facility), OSU, USA [103]	2200	Helium, MHTGR	2016-	Depressurised conduction cooldown and normal operation at 1:4 scale	850 (0.8)
Water-cooled RCCS, TAMU, USA [71]	-	Water, VHTR	In operation	RCCS natural convection and mixing	- (-)
Air and Water RCCS facilities, UW-M, USA [102]	30	HTR	In operation	RCCS heat removal and heat transfer	- (-)

Table 6: Thermal Hydraulic Test Facilities for Gas Cooled Reactors (GFR and VHTR)

Facility name, Organisation Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
CEDI, CIAE, China	450	Sodium	Refurb 2000-	Fuel assembly tests, including thermal shock (15 t sodium)	600 (0.9)
ESPRESSO, CIAE, China	450	Sodium	Refurb 2000-	Fuel rod tests under high temperature and thermal shock (3.2 t sodium)	600 (0.7)
FRIYG-I (Fast reactor fuel Rod sImulator hYdraulic test riG), CIAE, China	50	Water	2014-	Fuel rod simulator for flow distribution, pressure drop and FIV	95 (0.6)
SIPHON (Siphon broken device test rig), CIAE, China	50	Water	2003-	Test performance of Siphon breaker device	65 (0.25)
TSBS (Thermal hydraulic loops for Single-phase and Boiling two-phase Sodium), XJTU, China	170 90	Sodium	2010-	Single-phase loop for friction, heat transfer Two-phase loop for boiling experiments	500 (0.15) 1300 (0.12)
BACCARA (Banc de Caractérisation d'Assemblages de Réacteurs Avancés), CEA, France	115	Water	2015-	Fuel assembly vibration and cavitation	110 (1.5)
DIADEMO Na, CEA, France	40	Sodium, nitrogen	2013-	Test sodium/gas heat exchanger mock-ups	550 (0.05)
IRINA (Installation de Recherche pour l'Instrumentation en sodium NA), CEA, France	25	Sodium	Refurb 2011-	Test instrumentation, small components and thermal fatigue tests	550 (0.05)
NADYNE, CEA, France	300	Sodium	Due 2019	Development of elements such as fuel assemblies	700 (0.5)
NAIMMO, CEA, France	300	Sodium	Due 2019	Develop and qualify large components	580 (0.5)
NSET, CEA, France	10000	Sodium, nitrogen	Due 2019	Sodium-nitrogen heat exchanger development	530 (0.1)
PLATEAU (PLATeforme en EAU), CEA, France	550	Water	2014-	Thermal hydraulic tests on large scale mock-ups	60 (0.5)
DRESDYN (DRESden sodium facility for DYNamo and thermohydraulic studies), HZDR, Germany	1000	Sodium	2016-	Hydromagnetic dynamo effects, magnetorotational instability and two-phase flows (12 t sodium)	- (-)
NATAN (NATrium-versuchsANlage), HZDR, Germany	-	Sodium	1988-	Test materials and measurement techniques, two-phase flows and heat transfer	400 (-)

Facility name, Organisation Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
ALINA (kArlsruhe experiments with LI and NA free jet), KIT, Germany	120	Sodium	Stand by	Liquid metal jet, free surface studies	300 (0.1)
KASOLA (KArlsruhe SOdium LAboratory), KIT, Germany	800	Sodium	2015-	Turbulent and convective heat transfer, free surface flows	550 (0.1)
SOLTEC (SOdium Loop to TEst materials and Corrosion), KIT, Germany [83]	-	Sodium	2015-	Small scale material qualification and steel corrosion tests	950 (0.25)
500kW Sodium Loop, IGCAR, India	500	Sodium	Refurb 2015-	Sodium-sodium and sodium-air heat exchanger performance tests	550 (0.12)
LCTR (Large Component Test Rig), IGCAR, India	215	Water	1994-	Full scale testing of reactor components (80 t sodium)	550 (0.13)
SADHANA (SAfety Decay Heat Analysis in NAtrium), IGCAR, India	440	Sodium	2009-	Passive decay heat removal system tests at 1:22 scale	550 (0.12)
SAMRAT (Scaled model of reactor hydraulic), IGCAR, India	180	Water	2003-	Thermal hydraulic and FIV studies of reactor vessel at 1:4 scale	70 (1.0)
SGTF (Steam Generator Test Facility), IGCAR, India	5700	Sodium	2003-	Improve the design of once through SGs	525 (0.15)
Sub-assembly Hydraulic Test Rig, IGCAR, India	120	Water	1994-	Qualification of sub-assemblies, pressure drop devices and FIV tests	70 (1.6)
AtheNa (Advanced TecHnology Experiment sodium(Na) facility), JAEA, Japan	60000	Sodium	-	System and component demonstration, SG and pump development (240 t sodium)	570 (-)
CCTL (Core Component Thermal-hydraulic test Loop), JAEA, Japan	1000	Sodium	-	Heat transfer in fuel assembly, impact of porous blockage and jet mixing	625 (0.8)
PLANDTL (PLANt Dynamics Test Loop), JAEA, Japan	1000	Sodium	-	Test performance of decay heat removal systems, including natural circulation	625 (0.13)
STELLA-1 (Sodium inTegral Effect test Loop for safety simuLation and Assessment), KAERI, Korea	2500	Sodium	2014-	Sodium-sodium and sodium-air heat exchanger tests, including natural circulation	180 (0.2)
STELLA-2 (STELLA Phase 2), KAERI, Korea	2000	Sodium	2019-	Integral test facility for decay heat removal tests	- (-)

Facility name, Organisation Country of origin	Max Power (kWe)	Coolant, reactor type	Start date, Status	Test type	Max operating temperature °C (pressure MPa)
RIGADYN (RIGA mhd DYNamo test facility), IPUL, Latvia	200	Sodium	Refurb 2015-	MHD dynamo phenomenon, magnetic field-flow studies	300 (0.25)
TESLA (Experimental sodium loop), IPUL, Latvia	75	Sodium	2014-	Electromagnetic pump and equipment tests	500 (0.5)
ST-300 (Thermo-vacuum chamber for high temperature LM tests), IPUL, Latvia	300	Sodium, Lithium	1987-	High temperature tests of electromagnetic pumps	927 (-)
6B, ROSATOM, Russia	1200	Sodium, Na-K	Refurb 2011-	Core heat transfer and flow, heat exchanger tests	450 (0.6)
AR-1, ROSATOM , Russia	750	Sodium, Na-K	Refurb 2011-	Boiling in rod bundles, heat exchanger tests	950 (0.6)
B-2, ROSATOM , Russia	60	Air	Refurb 2011-	Heat exchanger velocity and pressure loss studies	50 (-)
Protva-1, ROSATOM , Russia	800	Sodium, Na-K	Refurb 2011-	Sodium contamination and purification tests	780 (0.06)
SGDI, ROSATOM , Russia	700	Air	Refurb 2011-	Hydrodynamics of reactor flow paths	50 (0.05)
SGI, ROSATOM, Russia	250	Water	Refurb 1996-	Thermal hydraulic characteristics of reactor and heat exchanger components	80 (2.5)
SPRUT, ROSATOM , Russia	2000	Na, Pb, Pb-Li	Refurb 2012-	Steam generator and heat exchanger thermal hydraulics	550 (0.6)
V-200, ROSATOM , Russia	150	Water	Refurb 2011-	Temperature and velocity in primary circuit under normal operation, transient and emergency cool down	95 (0.1)
ALEX (Argonne Liquid Metal EXperiment), ANL, USA	-	Sodium, lithium	1984-	Sodium plugging in narrow channels e.g. Printed Circuit Heat Exchanger	750 (13.8)
METL (Mechanisms Engineering Test Loop), ANL, USA	1000	Sodium	2016-	Test small and medium scale components and systems	650 (0.7)
SNAKE (S-CO_2, NA Kinetics Experiment), ANL, USA	-	CO_2, Sodium	2013-	Sodium-CO_2 interaction during gas injection through small hole	538 (20)
Sodium loops 1&2, UW-M, USA	2.4	Sodium	2009-	Materials, thermal striping, stratification and instrumentation development	650 (0.1)

Table 7: Thermal Hydraulic Test Facilities for Sodium-cooled Fast Reactors[54]

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[54] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
COMPLOT (COMPonent LOop Tests), SCK-CEN, Belgium	75	LBE, air	2014-	Full scale performance and hydraulic tests of MYRRHA core components (9 t LBE)	400 (0.9)
E-SCAPE (European SCAled Pool Experiment), SCK-CEN, Belgium	200	LBE, oil, air	2015-	1:6 scale model of MYRRHA to simulate thermal hydraulic phenomena (27 t LBE)	320 (0.4)
MYRRHABELLE (MYRRHA Basic sEt-up for Liquid fLow Experiments), VKI, Belgium	48	Water	2014-	Plexiglass 1:5 scale model of MYRRHA for thermal hydraulic experiments	30 (0.1)
CLEAR-S (China LEAd-based Reactor), INEST, China	3500	LBE	2016-	Test reactor equipment and TH phenomena in LBE pool (200 t LBE)	500 (2.0)
KYLIN II-TH FC (Thermal Hydraulics Forced Circulation Test Loop), INEST, China	300	LBE	2014-	Heat transfer, pressure drop and cladding temperature of rod bundles	400 (0.8)
KYLIN II-TH MC (Thermal Hydraulics Mixed Circulation Loop), INEST, China	500	LBE	2014-	Dynamic natural and forced circulation experiments in a test loop	500 (1.2)
KYLIN II-TH NC (Thermal Hydraulic Natural Circulation loop), INEST, China	24	LBE	2014-	Natural circulation tests with and without gas injection	500 (1.0)
ELEFANT (Experimental LEad FAcility for Neutron production Targets), HZDR, Germany	15	Lead	2002-	Test performance of heat exchangers, pumps and UDV measurement techniques	500 (0.6)
MINIPOT (Modular mINI reactor Pool for corrosion, Oxygen Transport and filtering), KIT, Germany	-	LBE	2014-	Oxygen transport, gas-metal interface, influence of water droplets	480 (0.1)
THEADES (THErmal hydraulics and Ads DESign), KIT, Germany	500	LBE	2002-	Rod bundle heat transfer, heat exchanger heat transfer	450 (1.0)
THESYS (Technologies for HEavy metal SYStems), KIT, Germany	20	LBE	2001-?	Instrumentation development, single heated rod heat transfer	400 (0.5)
CIRCE-ICE/HERO (CIRCulation Eutectic), ENEA, Italy	1000	LBE	2014-	Mixing and stratification in pool and fuel rod bundle heat transfer, HERO heat exchanger tests (90 t LBE)	120 (0.12)
CIRCE-SGTR (CIRCulation Eutectic), ENEA, Italy	30	LBE	2015-	LBE-water Steam Generator Tube Rupture test (90 t LBE)	350 (0.15)
HELENA (HEavy Liquid metal Experimental loop for advanced	250	Lead	2014-	Multipurpose facility for pump and valve tests, erosion /	480 (0.8)

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant	Start date, Status	Test type	Max operating temperature ºC (pressure MPa)
Nuclear Applications), ENEA, Italy				corrosion tests, heat transfer and thermal hydraulics	
LIFUS5 (LIthium FUSion), ENEA, Italy	90	LBE, Lead, PbLi	2000-	Interaction of water with liquid metal following SGTR event	220 (18)
NACIE-UP (NAtural CIrculation Experiment UPgrade), ENEA, Italy	235	LBE	2014-	Heat transfer and natural circulation behaviour of MYRRHA fuel assembly during Loss of Flow Accident	150 (0.8)
SIRIO (SIstema di RImozione della potenza di decadimento per reattori nucleari innOvativi), ENEA, Italy	70	Water	2016-	Investigate behaviour of decay heat removal system and ALFRED isolation condenser	450 (18)
SOLIDX (SOLIDification eXperiment facility), ENEA, Italy	5.3	Lead	2015-	Stagnant facility to study freezing and de-freezing	500 (0.12)
JLBL-3 (JAEA Lead-Bismuth flow Loop-3), JAEA, Japan	41	LBE	-	Beam window heat transfer tests	450 (0.5)
JLBL-4 (JAEA Lead-Bismuth flow Loop-4), JAEA, Japan	-	LBE	2010-	Development of ultrasonic flow and velocity systems, and oxygen sensors	500 (0.2)
TEF (Transmutation Experimental Facility), JAEA, Japan	250	LBE	-	Functional tests for LBE target and beam window	- (-)
TEF mock-up (Mock-up loop of TEF target), JAEA, Japan	-	LBE	2015-	Thermal hydraulic behaviour in mock-up scale, and ultrasonic flow meters	500 (0.2)
HELIOS (Heavy Eutectic liquid metal Loop for the Investigation of Operability and Safety of PEACER), NUTRECK, Korea	60	LBE	2005-	Forced circulation, natural circulation and materials corrosion tests	350 (0.1)
IPUL (Experimental Lead-Bismuth Loop), IPUL, Latvia	20	LBE	2013-	Small loop to test measurement techniques	450 (0.11)
B-2, IPPE, ROSATOM , Russia	60	Air	Refurb 2011-	Heat exchanger velocity and pressure loss studies	50 (-)
SGDI, IPPE, ROSATOM , Russia	700	Air	Refurb 2011-	Hydrodynamics of reactor flow paths	50 (0.05)
SGI, IPPE, ROSATOM, Russia	250	Water	Refurb 1996-	Thermal hydraulic characteristics of reactor and heat exchanger components	80 (2.5)
SPRUT, IPPE, ROSATOM , Russia	2000	Na, Pb, Pb-Li	Refurb 2012-	Steam generator and heat exchanger thermal hydraulics	550 (0.6)

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant	Start date, Status	Test type	Max operating temperature °C (pressure MPa)
V-200, IPPE, ROSATOM , Russia	150	Water	Refurb 2011-	Temperature and velocity in primary circuit under normal operation, transient and emergency cool down	95 (0.1)
TALL3D (Thermal-hydraulic ADS Lead-bismuth Loop with 3D flow test section), KTH, Sweden	80	LBE	2014-	Forced, natural and mixed circulation regimes, mixing and stratification in a pool and solidification phenomena	460 (0.7)
DELTA (DEvelopment of Lead Alloy Technical Applications), LANL, USA	50	LBE	2001- Standby	Long-term corrosion effects and thermal hydraulic studies	550 (0.7)

Table 8: Thermal Hydraulic Test Facilities for Lead-cooled Fast Reactors[55]

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[55] https://nucleus.iaea.org/sites/lmfns/Pages/default.aspx

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant	Status	Test type	Max operating temperature ºC (pressure MPa)
FFFER (Forced Fluoride Flow for Experimental Research), LPSC, France [115]	-	FLiNaK	2013-	Forced convection loop to study bubble separation process	900 (-)
SWATH (Salt at WAll: Thermal ExcHanges) facility, LPSC, France [115]	-	FLiNaK	Under design	Heat transfer, salt solidification interface and solidification on a cold wall	700 (-)
MSNCL (Molten Salt Natural Circulation Loop), BARC, India [116]	2	FLiNaK	In operation	Steady state and transient natural circulation in a loop	565 (0.15)
MAFL (Molten Active Fluoride Salt Loop) and MAF-Corr, BARC, India [117]	0.55	LiF–ThF_4	In operation	Natural circulation in a loop	750 (0.15)
DYNASTY (DYnamics of NAtural circulation for molten SalT internallY heated) loop, CIRTEN-PoliMi, Italy [118]	10	HITEC Heat Transfer Salt (HTS)	2016-	Transient natural circulation in a loop with and without internal heat generation	350 (-)
TNT (Tohoku-NIFS Thermofluid Loop), NIFS, Japan [119]	40	Heat Transfer Salt (HTS)	2002-	Heat transfer enhancement using packed-bed tube	600 (0.7)
LSTL (Liquid Salt Test Loop), ORNL, USA [120]	200	FLiNaK	In operation	Heat transfer, pump performance and instrument tests	700 (0.1)
Molten Salt Cell, ORNL, USA [120]	-	FLiNaK	In operation	Heat transfer and melting of molten salt in a pool	700 (-)
Forced convection FLiBe loop, ORNL, USA [120]	-	FLiBe	Under design	Forced convection heat transfer	- (-)
CIET (Compact Integral Effects Test), UCB, USA [121]	10	Dowtherm A oil	2015-	Transient forced and natural circulation response with two coupled loops	652 (-)
PBHTX (Pebble-Bed Heat Transfer eXperiment), UCB, USA [121]	10	Dowtherm A oil	In operation	Heat transfer in pebble-bed test section	90 (-)
LTDF (Low-Temperature DRACS Test Facility), UM, USA [114]	-	Water	In operation	Coupling between natural circulation loops	76 (1)
HT-FSTF (High-Temperature Fluoride Salt Test Facility), UM, USA [114]	-	FLiNaK	In operation	Coupling between natural circulation loops, salt heat transfer and pressure drop	722 (0.6)

Facility name, Organisation, Country of origin	Max Power (kWe)	Coolant	Status	Test type	Max operating temperature °C (pressure MPa)
Liquid Fluoride Salt Test Facility, UM, USA [114]	40	FLiNaK	In operation	Heat exchanger performance and hydrogen mass transfer	700 (-)
ARTIST (Advanced Reactor Technology Integral System Test) Facility, INL, USA [101]	200	Helium, FLiNaK, water	Under design	Heat transfer, FIV, multipurpose and multi-fluid test facility	325 (15)
Flibe natural convection loop, UW-M, USA [102]	10	FLiBe	In operation	Natural convection thermal hydraulics	- (-)

Table 9: Thermal Hydraulic Test Facilities for Molten Salt Reactors

ANNEX B - REACTORS UNDER DEVELOPMENT

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
FBNR, FURGS, Brazil	Fixed Bed Nuclear Reactor	Light water	72
ACPR50S, CGNPC, China Floating NPP	Advanced Customer-friendly Practicable Reliable	Light water	60
CNP-300, CNNC, China	CHASNUPP-1 small two loop	Light water	300
CAP200, SNERDI, China	China Advanced Passive PWR	Light water	220
SNP350, SNERDI, China	Advanced technology applied for CNP300	Light water	350
ATMEA1, ATMEA, France	ATMEA1	Light water	1100
EPR, AREVA, France	The Evolutionary Power Reactor	Light water	1770
Flexblue, DCNS, France	Subsea SMR	Light water	160
APWR, Mitsubishi, Japan	Advanced PWR	Light water	1538
APR+, Korea Hydro and Nuclear Power Company, Korea	Advanced Power Reactor Plus	Light water	1560
OPR1000, KEPCO/KHNP, Korea	Advanced Power Reactor	Light water	1050
APR1400, KEPCO/KHNP, Korea	Advanced Power Reactor 1400	Light water	1455
KLT-40S, OKBM, Russia	KLT-40S	Light water	35
VBER-300, OKBM, Russia	VBER-300	Light water	325
RUTA-70, RDIPE, IPPE, Russia	Integral pool-type heating reactor	Water	70 MW$_{th}$
SHELF, RDIPE, Russia	Underwater energy source	Light water	6
UNITHERM, RDIPE, Russia	Marine NPP	High purity water	6.6
ELENA, RRC KI, Russia	Direct conversion water-cooled reactor	Light water	0.068
VVER-1000, GIDOPRESS, Russia	VVER-1000 (V-466B)	Light water	1060
VVER-1200, GIDOPRESS, Russia	VVER-1200 (V-392M)	Light water	1170
VVER-1200, GIDOPRESS, Russia	VVER-1200 (V-491)	Light water	1170
VVER-1500, GIDOPRESS, Russia	VVER-1500 (V-448)	Light water	1560
VVER-300, GIDOPRESS, Russia	VVER-300 (V-478)	Light water	300
VVER-600, GIDOPRESS, Russia	VVER-600 (V-498)	Light water	600
VVER-640, GIDOPRESS, Russia	VVER-640 (V-407)	Light water	645
UK SMR, Rolls-Royce, UK	UK consortium PWR SMR	Light water	220
AP-1000, WEC, USA	Advanced Passive PWR	Light water	1200
AP-600, WEC, USA	Advanced Passive PWR	Light water	600
SMR-160, Holtec International, USA	Advanced PWR-type	Light water	160

Table 10: Pressurised Water Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
CAREM-25, CNEA, Argentina	Central Argentina de Elementos Modulares	Light water	27
ACP-100, NPIC, CNPE, CNNC, China	ACP-1000	Light water	125
ACPR100, CGNPC, China	Onshore NPP	Light water	140
IRIS, Internal Consortium	International Reactor Innovative & Secure modular	Light water	335
IMR, Mitsubishi, Japan	Integrated Modular Water Reactor	Light water	350
SMART, KAERI, Korea	System-integrated Modular Advanced ReacTor	Light water	100
ABV-6M, OKBM, Russia	Floating NPP	Light water	6
RITM-200, OKBM, Russia	Floating NPP	Light water	50
The SMART, GF Nuclear, UK	Developing KAERI SMART design	Light water	100
Westinghouse SMR, WEC, USA	Improves on AP-1000 plant	Light water	225
mPower B&W, USA (2 module plant)	Internal once-through steam generator	Light water	180 per module
NuScale, NuScale Power, USA (12 module plant)	NuScale Power Modular and Scalable Reactor	Light water	45 per module

Table 11: Integral Pressurised Water Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
KERENA, AREVA, France	KERENA	Light water	1290
RMWR, JAEA, Japan	Reduced-Moderation Water Reactor	Light water	1356
DMS, GE-Hitachi, Japan	Modular Simplified and Medium Small Reactor	Light water	300
KARAT-45, NIKIET, Russia	Independent cogeneration plant	Light water	45
KARAT-100, NIKIET, Russia	Integral type multi-purpose BWR	Light water	100
VK-300, RDIPE, Russia	Simplified integral BWR	Light water	250
ABWR, GE-Hitachi, USA	Advanced Boiling Water Reactor	Light water	1420
ABWR-II, GE-Hitachi, USA	Advanced Boiling Water Reactor II	Light water	1717
BWRX-300, GE-Hitachi, USA	SMR version of ESBWR	Light water	300
ESBWR, GE-Hitachi, USA	Economic Simplified Boiling Water Reactor	Light water	1600

Table 12: Boiling Water Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
ACR-1000, AECL, Canada	Advanced CANDU Reactor 1000	Light water	1165
EC6, AECL, Canada	Enhanced CANDU 6	Heavy water	740
AHWR300-LEU, BARC, India	Advanced Heavy Water Reactor	Light water	304
IPHWR-220, NPCIL, India	Indian 220MWe PHWR	Heavy water	220
IPHWR-700, NPCIL, India	Indian 700MWe PHWR	Heavy water	700

Table 13: Heavy Water Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
CANDU-SCWR, AECL, Canada	Thermal SCWR	Light water	1220
CSR1000, NPIC, China	Chinese Supercritical Water-Cooled Reactor	Light water	1000
HP-LWR, EURATOM, EU	High Performance LWR	Light water	1046
JSCWR, Toshiba, Japan	Japanese SCWR	Light water	1700
SCLWR-H, Tokyo, Japan	Supercritical LWR	Light water	1200
SCFBR-H, Tokyo, Japan	Supercritical Fast Breeder Reactor	Light water	700
Korean SCWR, KAERI, Korea	Solid moderator concept	Light water	1400
VVER-SCP, GIDOPRESS, Russia	Supercritical Pressure, single loop reactor plant	Light water	515
ChUWR, RDIPE, Russia	Thermal SCWR	Light water	1200
ChUWFR, RDIPE, Russia	Fast SCWR	Light water	1200
KP-SKD, RDIPE, Russia	Thermal pressure-tube SCWR	Light water	850

Table 14: SuperCritical Water Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
HTR-PM, INET, CNEC & Huaneng, China	High Temperature GCR, Pebble-Bed Module	Helium	210
ALLEGRO, EURATOM, EU	GFR demonstrator reactor	Helium	75 MW_{th}
GT-HTR300C, JAEA, Japan	Gas Turbine High Temperature Reactor	Helium	274
KAMADO FBR, CREIPI, Japan	KAMADO FBR	Carbon Dioxide	1000
NHDD, KAERI, Korea	Nuclear Hydrogen Development and Demonstration	Helium	200 MW_{th}
GT-MHR, OKBM, Russia	Gas turbine modular helium reactor	Helium	285
MHR-T reactor/Hydrogen production, OKBM, Russia	Multi-module plant for hydrogen production	Helium	4 x 205.5
MHR-100, OKBM, Russia	Modular helium reactor	Helium	87
PBMR-400, PBMR Pty Ltd, South Africa	Pebble Bed Modular Reactor	Helium	164
HTMR-100 SMR, STL, South Africa (4 module plant)	High Temperature Modular Reactor, pebble bed	Helium	35 per module
HTR-PM, DBD Ltd, UK	Working with INET to develop HTR-PM for UK	Helium	210
U-battery Developments Ltd, Urenco-led consortium, UK	Micro nuclear reactor with modular design	Helium	4 per unit
EM2, General Atomics, USA	Energy Multiplier Module	Helium	240
Prismatic HTR, General Atomics, USA	Prismatic Modular High Temperature GCR	Helium	150
GT-MHR, General Atomics, USA	Gas turbine modular helium reactor	Helium	150
SC-HTGR, AREVA, USA	Steam Cycle High Temperature Gas-Cooled Reactor	Helium	272
StarCore HTGR, StarCore Nuclear, Canada	StarCore Module High-Temperature Gas	Helium	120
USNC MMR-5&10, Ultra Safe Nuclear Corporation, USA	Micro-Modular Reactor	Helium	5-10
Xe-100, X-energy, USA	Pebble bed high temperature gas-cooled reactor	Helium	35

Table 15: Gas-cooled Reactor Designs (HTGR and GFR)

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
CEFR, CNEIC, China	China experimental fast reactor	Sodium	20
CFR-600, CIAE, China	China Fast Reactor 600	Sodium	600
ASTRID, CEA and partners, France	Advanced Sodium Technological Reactor for Industrial Demonstration	Sodium	600
PFBR-500, IGCAR, India	Prototype fast breeder reactor	Sodium	500
FBR-1 & 2, IGCAR, India	Fast Breeder Reactors 1 & 2	Sodium	500
4S, Toshiba, Japan	Super-Safe, Small and Simple Reactor	Sodium	10
JSFR, JAEA, Japan	Japan Sodium-cooled Fast Reactor	Sodium	750
PGSFR, KAERI, Korea	Prototype Gen IV Sodium-cooled Fast Reactor	Sodium	150
BN-1200, OKBM, Russia	BN-1200	Sodium	1220
MBIR, NIKIET, Russia	Multipurpose fast-neutron research reactor	Sodium	60
ARC-100, Advanced Reactor Concepts LLC (ARC), USA	Sodium fast nuclear reactor	Sodium	100
PRISM, GE-Hitachi, USA	Power Reactor Innovative Small Reactor	Sodium	311
TWR-P, TerraPower, USA	Travelling Wave Reactor-Prototype	Sodium	600

Table 16: Sodium-cooled Fast Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
MYRRHA, SCK-CEN, Belgium	Multi-purpose hYbrid Research Reactor for High-tech Applications	LBE	100 MW_{th}
LEADIR-PS100, NNI, Canada	LEAD-cooled Integral Reactor - Passively Safe	Lead	36
CLEAR-I, INEST, China	China LEAd-based Research Reactor	LBE	10 MW_{th}
ALFRED, Ansaldo Nucleare, EU	Advanced Lead Fast Reactor European Demonstrator	Lead	125
ELFR, Ansaldo Nucleare, EU	European Lead Fast Reactor	Lead	630
PEACER, SNU, Korea	Proliferation-resistant Environment-friendly Accident-tolerant Continuable and Economical Reactor	LBE	300
BREST-OD-300, RDIPE, Russia	BREST-OD-300	Lead	300
SVBR-100, AKME Engineering, Russia	SVBR-100	LBE	101
ELECTRA, KTH, Sweden	European Lead Cooled Training Reactor	Lead	0.5 MW_{th}
SEALER, LeadCold, Sweden	Swedish Advanced Lead Reactor	LBE	40
G4M, Gen4 Energy, USA	Gen4 Module	LBE	25
LFR-AS-200, Hydromine Nuclear Energy, USA	Amphora Shaped, Lead fast reactor	Lead	200
SSTAR, Lawrence Livermore team, USA	Small Secure Transportable Autonomous Reactor	Lead	20
Westinghouse LFR, Westinghouse, USA [122]	Westinghouse Lead-cooled Fast Reactor	Lead	400

Table 17: Lead-cooled Fast Reactor Designs

Acronym, Organisation, Country	Full name	Coolant	Power (MWe)
IMSR-400, Terrestrial Energy, Canada	Integral Molten Salt Reactor-400	Flouride salts	192
TMSR-LF, SINAP, China	Thorium-breeding molten-salt reactor, liquid fuel	FLiBe	1000
TMSR-SF, SINAP, China	Thorium-breeding molten-salt reactor, solid fuel	FLiBe	1000
MSTW, Seaborg Technologies, Denmark	Molten Salt Thermal Wasteburner	Flouride salts	115
MSFR, CNRS, France	Molten Salt Fast Reactor	Flouride salts	1500
ThorCon, Martingale, International Consortium	Thorcon Molten Salt Reactor	Flouride salts	250 per module
MSR-FUJI, ITMSF, Japan	Molten Salt Reactor-FUJI	FLiBe	200 per module
MOSART, Kurchatov Institute, Russia	Molten Salt Actinide Recycler Transmuter	Flouride salts	1100
SSR-U, Moltex Energy, UK (8 module plant)	Stable Salt Reactor	Flouride salts	300 (37.5 per module)
Mk1 PB-FHR, UCB, USA	Mark 1 Pebble-Bed Fluoride-Salt-Cooled High Temperature Reactor	FLiBe	100
MCFR, TerraPower, USA	Molten Chloride Fast Reactor	Chloride salts	1150
MCSFR, Elysium, USA	Molten Chloride Salt Fast Reactor	Chloride salts	1000
SmAHTR, ORNL, USA	Small fluoride salt-cooled Advanced High Temperature Reactor	Flouride salts	125 per module
LFTR, Flibe Energy, USA	Liquid-Fluoride Thorium Reactor	FLiBe	250 per module
Transatomic Power, TAP, USA	Single-fluid MSR	Flouride salts	550

Table 18: Molten Salt Reactor Designs

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Originating Office: FRAZER-NASH CONSULTANCY LIMITED
The Cube, 1 Lower Lamb Street, Bristol, BS1 5UD
T: +44 (0)117 9226242   F: +44 (0)117 9468924   W: www.fnc.co.uk

Frazer-Nash Consultancy Ltd
The Cube
1 Lower Lamb Street
Bristol
BS1 5UD

T +44 (0)117 9226242
F +44 (0)117 9468924

www.fnc.co.uk

Offices at:
Bristol, Burton-on-Trent, Dorchester,
Dorking, Glasgow, Plymouth, Warrington
and Adelaide